1.0 GENERAL INFORMATION

1.1 Introduction

This Safety Analysis Report (SAR) presents the evaluation of a Type B(U) spent fuel transport packaging developed by Transnuclear, Inc. and designated the TN-40. This SAR describes the design features and presents the safety analyses which demonstrate that the TN-40 complies with applicable requirements of 10 CFR 71 [1]. The format and content of this SAR follow the guidelines of Regulatory Guide 7.9 [2].

The TN-40 is a dual purpose cask intended for both storage and transport. The TN-40 is currently licensed for storage at the Prairie Island Nuclear Generating Plant (Docket No. 72-0010). A separate storage SAR was submitted by Prairie Island in support of the storage license application. It addresses the safety related aspects of storing spent fuel in TN-40 casks in accordance with 10 CFR 72 [3].

The packaging is to be licensed for a one-time use. That is, for shipment of the spent fuel it contained during storage. This one-time use is defined to include any sequence of shipments as long as the spent fuel originally loaded has not been removed. The packaging is intended to be shipped as exclusive use. The Criticality Safety Index (CSI) for nuclear criticality control for the TN-40 cask is determined to be zero (0) in accordance with 10 CFR 71.59 [1]. See Chapter 6 for details of this determination.

Transnuclear, Inc. has an NRC approved quality assurance program (Docket Number 71-0250) which satisfies the requirements of 10 CFR 71 Subpart H [1].

1.2 Package Description

1.2.1 Packaging

The TN-40 packaging will be used to transport 40 PWR undamaged fuel assemblies with or without fuel inserts. In its transport configuration, the TN-40 packaging consists of the following components:

- A basket assembly which locates and supports the fuel assemblies, transfers heat to the cask inner shell, and provides sufficient neutron absorption to satisfy nuclear criticality requirements.
- A containment vessel including a closure lid and metallic seals which provides radioactive materials containment and maintains an inert gas atmosphere.
- A thick-walled, forged steel gamma shield shell, bottom shield and lid shield plate provide shielding that surrounds the containment vessel.
- A radial neutron shield surrounding the gamma shield shell which provides additional radiation shielding. The neutron shielding is enclosed in a steel outer shell.

A 6.0 in. thick shield plate (SA-105 or SA-516, Grade 70) is also welded to the inside of the lid outer plate (drawing 10421-71-4).

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The resin compound is cast into long, slender aluminum alloy containers. The total radial thickness of the resin and aluminum is 4.50 in. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell (SA-516, Grade 55 or equivalent) constructed of two half cylinders. In addition to serving as resin containers, the aluminum containers provide a conduction path for heat transfer from the gamma shield shell to the outer shell. A pressure relief valve is mounted on top of the resin enclosure to limit the internal pressure increase that may be caused by heating of the resin enclosure for hypothetical accident conditions.

The resin material is an unsaturated polyester cross-linked with styrene, with approximately 50 weight % mineral and fiberglass reinforcement. The components are polyester resin, styrene monomer, alpha methyl styrene, aluminum oxide, zinc borate, and chopped fiberglass which produce the elemental resin composition shown below.

Element	wt%
H	5.05
B	1.05
C	35.13
Al	14.93
O + Zn (balance)	43.84

The resin used for the radial neutron shield is a proprietary formulation that has been utilized for the TN-40, TN-32, and TN-68 casks which have been licensed for storage. The TN-68 cask is also licensed for transport. Information on the resin has been provided to the NRC in support of their license applications. Appendix 9A of the TN-68 storage FSAR provides information on the resin. Information from Appendix 9A of the TN-68 storage FSAR is provided below.

The resin material is an unsaturated polyester crosslinked with styrene, with about 50 weight % mineral and fiberglass reinforcement. The components are polyester resin, styrene monomer, alpha methyl styrene, aluminum oxide, zinc borate, and chopped fiberglass.

Thermal Stability

Thermal aging tests on a material with the same components in slightly different proportions have been performed by Transnucleaire, Paris (TNP). The tests by TNP evaluate weight loss and off-gassing at 260 °F (125 °C) and 311 °F (155 °C). The maximum normal temperature in the TN-40 radial neutron shield is 233 °F (111 °C) at the beginning of storage per Chapter 3 of the TN-40 SAR and for NCT the resin temperature is 229 °F. An exponential weight loss occurs that rapidly approaches a maximum value. After 106 hours, the weight loss is about 1.0%, and extrapolation of the

results indicates maximum weight loss of about 1.3%. This effect diminishes rapidly with decreasing temperature. An analysis of the gas released from a sample heated from 25 to 125 °C over one hour shows it to be 99.9% styrene.

These results obtained with small samples (50 mm thick x 50 mm dia) are conservative with respect to the material in a larger enclosed form such as the TN-40 radial neutron shield, where volatile constituents must diffuse through a much greater distance to be released.

Radiation Stability

The European Organization for Nuclear Research (CERN) has published a compilation of its own testing and of prior published data on the radiation resistance of various materials. The data show that while unfilled polyester has poor radiation resistance, both mineral- and glass-filled polyester, such as used in the TN-40 radial neutron shield, are among the most radiation-resistant of thermosetting resins.

There are over 100 TN-40, TN-32, and TN-68 casks currently in storage at various ISFSI sites in the U.S. Periodic inspections and dose rate measurements of the casks in storage have not indicated any evidence of deterioration of the neutron shielding.

Furthermore, prior to transport of the TN-40 packaging, dose rate measurements must be performed to demonstrate that they meet the 10CFR71.47 criteria. This assures that the resin material has retained adequate properties to meet transport requirements.

The structural analysis of the TN-40 cask shielding is presented in Chapter 2.

Noncontainment welds are inspected in accordance with the NDE acceptance criteria of ASME B&PV Code Subsection NF [8].

1.2.1.3 Impact Limiters

Top (front) and bottom (rear) impact limiters, shown on drawings 10421-71-2 and -40 through -44, form a part of the TN-40 packaging. The impact limiters are attached to each other using 13 tie rods and to the cask by bolt attachment brackets welded to the outer shell in eight locations (four bolting locations per impact limiter). The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells (A-240, Grade 304) that maintain a dry atmosphere for the wood and confine the wood when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement.

The impact limiters have an outside diameter of 144 in., and an inside diameter of 92 in. to accommodate the cask ends. The bottom limiter is notched to fit over the lower trunnions. The impact limiters extend axially 37.75 in. from either end of the cask, and overlap the sides of the cask by 12.25 in.

1.2.3 Contents of Packaging

The characteristics of the contents of the TN-40 packaging are limited to the following:

a. Fuel shall be unconsolidated;

b. Fuel shall only have been irradiated at the Prairie Island Nuclear Generating Plant Unit 1 cycles 1 through 16 or Unit 2 cycles 1 through 15;

c. Fuel shall be limited to fuel types:

   i. Westinghouse 14X14 Standard,

   ii. Exxon 14X14 Standard (includes high burnup standard),

   iii. Exxon 14X14 TOPROD, and

   iv. Westinghouse 14X14 OFA;

d. Fuel may include burnable poison rod assemblies (BPRAs) provided:

   i. the BPRA has cooled for ≥ 18 years,

   ii. the cask average cumulative burnup of the fuel assembly(s) where the BPRA(s) resided during reactor operation shall be ≤ 30,000 MWd/MTU;

e. Fuel may include thimble plug assemblies (TPAs) provided:

   i. the TPA has cooled for a minimum of 13 years,

   ii. the cask average cumulative burnup of the fuel assembly(s) where the TPA(s) resided during reactor operation shall be ≤ 125,000 MWd/MTU;

f. The combined weight of a fuel assembly and any BPRA or TPA shall be < 1330 lbs;

g. The combined weight of all fuel assemblies, BPRAs, and TPAs in a single cask shall be < 52,000 lbs;

h. The number of assemblies in the container shall be ≤ 40;

i. The initial enrichment shall be ≤ 3.85 weight percent U-235;

j. The assembly average burnup shall be ≤ 45,000 MWd/MTU;

k. The assembly average burnup shall be ≥ the loading curve shown in Figure 1-2;

l. The cooling time prior to transport shall be in accordance with Table 1-2; and

m. The fuel assemblies shall not be Unit 1 Region 4 fuel assemblies, i.e. assemblies identified as D-01 through D-40.

n. Physical data:

Max unirradiated Length (in.)	*161.3*
Max Width (in.)	*7.763*
No of Fueled Rods	*179*
Clad Material	*Zr-4*
Guide Tube #	*16*
Instrument Tube #	*1*
Maximum MTU/assembly	*0.410*

o. The fuel shall not be DAMAGED FUEL ASSEMBLY.

DAMAGED FUEL ASSEMBL*Y is a* spent nuclear fuel assembl*y that*:

- *is a partial fuel assembly, that is, a fuel assembly from which fuel pins are missing unless dummy fuel pins are used to displace an amount of water equal to that displaced by the original pins; or*
- *has known or suspected to have structural defects or gross cladding failures (other than pinhole leaks) sufficiently severe to adversely affect fuel handling and transfer capability.*

30 Foot Drop Orientation	g Load Measured by Testing *(See Table 2.10.9-1 of Appendix 2.10.9)*
90° End Drop	*54* g Axial
0° Side Drop	*51* g Transverse
CG Over Corner Drop	34 g Axial
20° Slap Down (Second Impact)	*58* g[(1)]

(1) The top/bottom ends of the cask body are enclosed by the impact limiters. There are no accelerometers located either on the impact limiters or on the cask body inside of the impact limiters. As shown on Figure 2.10.9-2, the accelerometers are located next to the impact limiters. These locations are coincident to the top/bottom ends of the basket sections. The g loads measured at these locations represent the maximum combined transverse and rotational g load for the basket structural analysis due to the slap down drop case.

For the slap down testing case, the measured testing g load did not separate the transverse g load and rotational g load. However, since the slap down combined g loads from the ADOC run are higher than the test combined g loads, the g loads estimated from the ADOC are used as reference point.

Figure 2-3 shows a free body diagram of the cask based on ADOC output during the first impact. The maximum transverse g load is 39g and maximum rotational g load is 51g at the first impact end. The maximum combined transverse and rotational g load is 90g as shown in Appendix 2.10.8, Table 2.10.8-3. Figure 2-4 depicts the 20° slapdown first impact acceleration diagram.

Figure 2-5 shows a free body diagram of the cask based on ADOC output during the second impact, the maximum transverse g load is 27g and maximum rotational g load is 49g at the second impact end. The maximum combined transverse and rotational g load is 76g as shown in Appendix 2.10.8, Table 2.10.8-3. Figure 2-5 depicts the 20° slapdown second impact acceleration diagram.

For the slap down drop case, the second impact (combined transverse g load and rotational g load) will have more severe impact to the components than the first impact. Therefore the reported g load for the slap down is based on the second impact. These combined g loads are higher than the side drop g loads, therefore, for the basket and fuel rod side drop analyses, the applied g load must bound both side drop and slap down g loads.

Based on the ADOC 20° Slap Down run, the first impact force is higher than the second impact force; the structural analysis of the cask body is based on the first impact loads. As described in Appendix 2.10.1, the transverse, axial and rotational accelerations are applied as inertial loads in the transverse and axial directions and the rotation acceleration is applied at the cask body CG. As also described in Appendix 2.10.1, load case 18 (20° slapdown impact on lid end) and load case 19 (20° slapdown impact on bottom end), the rotational acceleration input to the model to balance the forces are much higher than the ADOC estimated accelerations. This is due to internal load distribution inside the cask inner surface. The pressure due to the weight of internals is input as a cosine varying pressure around the radial portion of the cavity (uniform through the cavity length). This input methodology requires higher rotational acceleration to balance the force compared to using a load distribution based on linear through the cavity length (maximum at the impact end and minimum at other end). Both

methodologies are analyzed, and the stresses based on uniform through the cavity length are shown to be bounding. Therefore these are reported in the SAR.

The effect of the low temperature (-20 ^{o}F) on the tested impact limiter is not available due to lost test data. However, based on the similar design, TN-68 impact limiter testing [30], chilling the impact limiter wood (-20 ^{o}F) will increase the g load by 15% to 20%. This is based on the measured g loads from the TN-68 testing that are approximately 75g (-20 ^{o}F) and the maximum g loads predicted by ADOC are approximately 66g (room temperature) (75/66 ≈ 14%). An increase of *15*% in the accelerations from bounding ADOC and testing g loads will bound the low temperature effect on the wood properties.

The following table summarizes the baseline g loads to bound the g load for the package component structural evaluations.

Baseline g Loads for Cask Body Structural Analyses

30 Foot Drop Orientation	***Bounding g Loads Source***	***(-20 ^{o}F) Low Temperature Factor***	***Bounding Baseline g Loads Used for Cask Body Structural Analyses***
90° end drop	*54 g axial (testing)*	*1.15 x 54 = 62*	*68 g (axial)*
0° side drop	*51 g transverse (ADOC)*	*1.15 x 51 = 59*	*68 g (transverse)*
CG over corner drop	*34 g axial (testing)* *14 g transverse (ADOC)*	*1.15 x 34 = 39* *1.15 x 14 = 16*	*41 g (axial)* *18 g (transverse)*
20° slap down (second impact)	*27 g transverse (ADOC)* *2 g axial (ADOC)* *49 g rotational (ADOC)(1)*	*1.15 x 27 = 31* *1.15 x 2 = 2.3* *1.15 x 49 = 56*	*47 g (transverse)* *26 g (axial)* *61 g (rotational)*

(1) The maximum combined g load is 76 g (27 g + 49 g) at the top end of the cask body (at the outer surface of the cask lid). For the maximum g load at locations other than the top end of the cask body (such as at the top end of the basket section), the rotational g load needed to be ratioed by the distance to the package CG and then added the transverse g load (27 g).

The following approaches are used for the component structural evaluations to demonstrate that the structural integrity of the components is maintained.

A. *The bounding g loads* multiplied by appropriate dynamic load factors *and factors due to low temperature effect* are used for the basket side drop and end drop structural evaluations. The g loads are calculated in the following table.

Baseline g Loads for Basket Structural Analysis

Drop Orientation	***Bounding g Load Source***	***Load Factor***	***g Load Used in the Analysis***
End drop	*54 g (testing)*	*54 x 1.08(1) x 1.15(2) = 67 g*	*75 g*
Side drop	*51 g (ADOC)*	*51 x 1.08(1) x 1.15(2) = 63 g*	*75 g side drop analysis bounds both side drop and slap down drop*
Slap down	*58 g(3)*	*58 x1.08(1) x 1.15(2) = 72 g*	

(1) Dynamic load factor, see Appendix 2.10.6.
(2) Wood property low temperature effect.
(3) The slap down bounding g load source is taken from the testing results, this is based on:
a. The testing g load is measured at the top end of the basket section and ADOC results are reported at the top end of the cask body. For the ADOC run, in order to get the maximum combined g load at the top end of the basket section, it is required to ratio the rotational g load from the top end of the cask back to the top end of the basket section and then combined with the transverse g load to get the maximum combined g load. The basket top end g load measured from the testing is already a combined g load (transverse and rotational) and therefore is more direct and appropriate for the basket analysis.
b. The impact limiters used for the testing are fabricated to closely meet the full size geometry and wood properties requirements. Therefore the results from the testing are considered more realistic then the g load calculated by using

ADOC (certain assumptions are used in the program, such as % of wood backed up by the cask body during the drop, wood locking strain and modulus, etc.).

B. *The bounding g loads* multiplied by appropriate dynamic load factors *and factors due to low temperature effect* are used for the fuel rod side drop and end drop structural evaluations. The g loads are calculated in the following table.

Baseline g Loads for Fuel Rod Structural Analysis

Drop Orientation	***Baseline g Load***	***Load Factor***	***g Load Used in the Fuel Rod Analysis***
End drop	*(1)*	*(1)*	*(1)*
Side drop	*51 g*	*51 x 1.08(2) x 1.15(3) = 63 g*	*75 g side drop analysis bounds both side drop and slap down drop*
Slap down	*58 g(4)*	*58 x1.08(2) x 1.15(3) = 72 g*	

(1) The response curve from the TN40 1/3 scale end drop test (Figure 2.10.9-24) was used. Since the test model was 1/3 of the original size, all of the acceleration values are scaled by 1/3 and all of the times are scaled by a factor of 3. Furthermore, for the -20°F temperature effect, the acceleration values were increased by 15% and the time values were decreased by 15%. The response time history used in the analysis is shown on Figure 2.10.7-5.
(2) Dynamic load factor, see Appendix 2.10.7.
(3) Wood property low temperature effect.
(4) See note 3 on Table "Baseline g Loads for Basket Structural Analysis".

C. Cask body structural analysis
   a. The stresses in the cask body are calculated for the end drop onto bottom (rear) end, end drop onto lid (front) end, side drop, CG over corner drop on bottom (rear) end, CG over corner over lid (front) end, 20° slap down impact on lid (front) end, and 20° slap down impact on bottom (rear) end.
   b. The baseline g loads established from the testing, ADOC runs, and -20°F temperature effect as listed in the previous tables are used for the cask body structural analyses.
   c. The ADOC analysis provides separate transverse, axial and rotational g loads for the slap down drop orientation and has been directly used for the ANSYS finite element model inputs. Therefore, for consistence, g loads calculated from ADOC are used for all the above drop orientation structural evaluations (including end, side, and CG over corner drops) as described in Appendix 2.10.1.
   d. Elastic analyses are used for all the cask body drop analyses in Appendix 2.10.1. Therefore, in order to calculate the stresses due to the baseline g loads, the load combinations as described in Table 2-17 are performed as follows:
      - Calculated load combination stresses based on the individual load stresses calculated in Appendix 2.10.1.
      - Calculated the new load combined stresses by ratioing the baseline g values vs. g values used in the calculations.

The nodes between the steel tubes including the intermediate aluminum plates are coupled together in the out-of-plane direction so that they will bend in unison under surface pressure or other lateral loading to simulate through the thickness support provided by Boral® plates. The aluminum plates are coupled together at their intersection. The fusion welds, connecting the fuel compartments and plates, are modeled with short pipe elements connected at each end to adjacent fuel compartment boxes in all directions. The nodal couplings are shown in Figure 2.10.5-22. Furthermore, a sensitivity study on the effect of the nodal couplings was performed and is provided in Section 2.10.5.5.

B. Material Properties and Design Criteria

The stainless steel boxes are constructed from SA-240, Gr. 304 stainless steel. The aluminum plates, outer plates and basket periphery plates are constructed from SB-209, 6061-T651 aluminum alloy. A bilinear stress-strain curve for SA-240 Type 304 stainless steel and SB-209 Type 6061-T651 aluminum alloy ($E_P/E = 0.05$) is used for the basket plates.

Data from a stress-strain curve for SA-240 Type 304 was taken from NUREG/CR 0481. This data used as a basis for the 5% strain-hardening rate shown in the bilinear stress-strain curve used in the basket analysis. Using a 5% strain-hardening rate which is greater than those resulting from this data is conservative because it will result in higher stresses.

For the aluminum SB-209 Type 6061-T651, Kaufman, J. Gilbert, "Properties of Aluminum Alloys: Tensile, Creep, and Fatigue Data at High and Low Temperatures," gives elongations of 17 – 70% with associated strain hardening rates less than 5%. In the case of aluminum, also, using a 5% strain-hardening rate is conservative because it will yield higher stresses.Note that all P_m limits ($0.7S_u$) are below S_y, therefore the strain-hardening rate has no effect on P_m stresses.

Table 2.10.5-1 lists the material properties used in all analyses of the TN-40 fuel basket. Tables 2.10.5-2 and 2.10.5-3 summarize the stress criteria for the NCT and HAC events respectively.

C. Side Drop Loading Conditions

The basket structure is analyzed for 0°, 45° and 90° azimuth side drops. Due to the basket structure symmetry, these orientations of side drops are assumed to envelop all other possible drop orientations.

A fuel assembly weight of 1,300 lb. is used in the analysis. A uniform fuel weight distribution is assumed over 144 inches, which is the active fuel length. An 8.0 inch sector of the basket assembly is modeled. The weight of the Boral® plates is accounted for by increasing the density of the aluminum plates. The Boral® plate's stiffness is conservatively neglected in the analysis.

The buckling loads are summarized in the table below. Figure 2.10.5-34 shows the displacement at the last converged load step (buckling load) for the bounding case, 0° side drop.

Basket Side Drop Orientation	**Buckling Load for Node Coupling Model(2) (g)**	**Buckling Load for Contact Element Model (g)**
0˚	145.44	95.2
30˚	88.54	96.6(1)
45˚	92.54	97.6(1)
90˚	115.15	96.6

Notes:

(1.) Solution failed to converge due convergence issue, conservatively the last converged was used as the buckling load

(2.) Results from Section 2.10.5.3

Changing node couplings to contact impacts the buckling loads of the 0˚ and 90˚ side drops, however the buckling loads of the bounding cases, 30˚ and 45˚, are higher. In conclusion, the analysis with couples remains bounding.

2.10.5.5.2 Fuel Boxes Initial Imperfection Sensitivity Analysis

The impact of initial imperfection in the fuel boxes was studied by applying an initial imperfection in the finite element model described in Section 2.10.5.5.1. Figure 2.10.5-35 shows the initial imperfection applied. Since 0˚ and 90˚ side drops are going to be affected by the initial imperfection, only the 0˚ orientation is analyzed in this study. All conclusions from the 0˚ side drop case would be applicable to the 90˚ side drop also.

The boundary and loading conditions are described in Section 2.10.5.5.1 and were unchanged. Buckling analyses (as described in Section 2.10.5.3) was performed for the 0° drop orientation.

The buckling load for 0˚ side drops increases to 96.1g from 95.2g, which is marginal. Displacement pattern, at the last converged sub-step (buckling load) is shown in Figure 2.10.5-36.

2.10.5.5.3 Tests Performed to Support Design of the TN-40 Basket

<u>Description of the Test</u>

In revision 1 of the Prairie Island ISFSI (Appendix 4C of SAR, Docket 72-10) [4], tests were performed to support the design of the TN40 basket. Compression tests were performed to determine the basket panel strength when loaded in the 8.05 in. direction (to simulate loading of a bottom outer radial panel of a basket during side impact). Each test panel was 8.05" long and 24" wide (the basket axial direction). Panels were tested with a weld spacing of 6", 8", and 12" in the axial direction. The actual weld spacing is

8". Panels were tested at room temperature and at elevated temperature. The loaded edges of the panel were hinged so that no panel edge rotational restraint would be provided. This results in conservatively low panel failure loads since, in the actual basket arrangement, there is some edge rotational restraint at the corner of the stainless steel box sections and at the aluminum plate continuation into adjacent panels. The edges were loaded through a lubricated brass insert free to rotate in the test fixture.

Test Results

The test results are summarized in the following table for easy reference.

Test No.	Weld Spacing (in.)	Temperature (^oF)	Maximum Load (lb, Total)	Maximum Unit Load (lb/in.)
A1	5 x 12	Room temp.	340,000	14,166
B1	5 x 8	Room temp.	332,000	13,833
C1	5 x 6	Room temp.	320,000	13,333
A2	5 x 12	529 ^oF	253,000	10,542
B2	5 x 8	405 ^oF	267,250	11,135
C2	5 x 6	365 ^oF	261,500	10,896

Correction Factor of the Test Compressive Load

In Appendix 4C [4], the plate thickness used for the test specimens is 0.110" for stainless steel plate and 0.256" for the aluminum plate. A correction factor is applied since the nominal plate thicknesses are 0.1 in and 0.25 in for fuel compartment and aluminum plates, respectively.

$$\text{Correction Factor, } F = \frac{\left({Lb_1^3}/{12} + {Lb_2^3}/{12} + Ah_1^2 + Ah_2^2 \right)}{\left({Lb_3^3}/{12} + {Lb_4^3}/{12} + Ah_3^2 + Ah_4^2 \right)}$$

$$= \frac{\left({0.256^3}/{12} + {0.11^3}/{12} + (0.256)({0.256}/{2})^2 + (0.11)(0.256 + {0.11}/{2})^2 \right)}{\left({0.25^3}/{12} + {0.1^3}/{12} + (0.25)({0.25}/{2})^2 + (0.1)(0.25 + {0.1}/{2})^2 \right)}$$

$$= \frac{0.016343}{0.014292} = 1.1435$$

Where b_1, b_2, h_1, and h_2 are plate thickness and neutral axis offset for the test specimen aluminum and stainless steel, respectively. Similarly b_3, b_4, h_3, and h_4 are the plate

thickness and neutral axis offset for the TN40 basket panel and A = length (1") x thickness.

The minimum test buckling load is 10,542 lb/in. (at 529 ^{o}F). Applying the correction factor, the calculated test buckling load based on the nominal plate thicknesses gives:

Test buckling load = 10,542 / 1.1435 = 9,219 lb/in.

Compressive Load Calculated by the ANSYS Model

To calculate the compressive load during a 75 g accident side drop event, a finite element model was created in ANSYS [1]. The test setup in the Appendix 4C has the basket panel as pin-pin connection, therefore TN40 basket panels are modeled with pin-pin supports. A 90° drop orientation, which results in the maximum compressive load in the bottom-most fuel compartment, is analyzed in the calculation.

The purpose of the analysis is to get the maximum compressive load in the TN40 basket panels. To simplify the analysis the sandwich (aluminum + steel) panels of the basket are modeled as a single plate using ANSYS Shell43 elements with 0.5" thickness. Furthermore, the panels are modeled in such a way that there is no moment transfer through any of the fuel compartments. The basket is modeled as an 8" section in the axial direction. Figure 2.10.5-37 shows the finite element model of the representative TN40 basket.

The basket is supported at the rail locations appropriate to the drop orientation. Symmetric boundary conditions are applied to the cut boundary locations.

Elastic material properties are used. The density of the material is adjusted such that the weight of the single plate model is identical to the basket weight.

The load resulting from the fuel assembly weight was applied as pressure on the fuel compartment plates of the basket.

A linear elastic side drop analysis of the basket is performed using ANSYS. The maximum compressive force in the bottom most fuel compartment wall is calculated to be 42,530 lb. Since the basket modeled as 8" section, therefore the unit load in the fuel compartment wall is 42,530/8 = 5316.25 lb/in.

Conclusion

The maximum compressive load for 75 g accident side drop load is 5,316 lb/in. The buckling load from the test specimen is 9,219 lb/in. Therefore, the factor of safety against buckling of the panel is 1.73 (9219/5316). Based on the test, it concluded that the basket can withstand up to 130 g (75 x 1.73) before reach the buckling load.

2.10.5.6 References

1. ANSYS Engineering Analysis System User's Manual, *Releases* 8.0 *and 10.0*.
2. ASME Boiler and Pressure Vessel Code, 1989, Section III, Subsection NB, NF & Appendices; Section VIII, Divs I &2.
3. "Aluminum Standards and Data," The Aluminum Association, Inc., 1976.
4. *Prairie Island ISFSI Technical Specification and Safety Analysis Report, Revision 1, 1991.*
4. *"An Assessment of Stress-Strain Data Suitable for Finite Element Elastic-Plastic Analysis of Shipping Containers" NUREG/CR-0481, SAND77-1872.*
5. *Kaufman, J. Gilbert, Properties of aluminum alloys: tensile, creep, and fatigue data at high and Low Temperatures, 1999.*

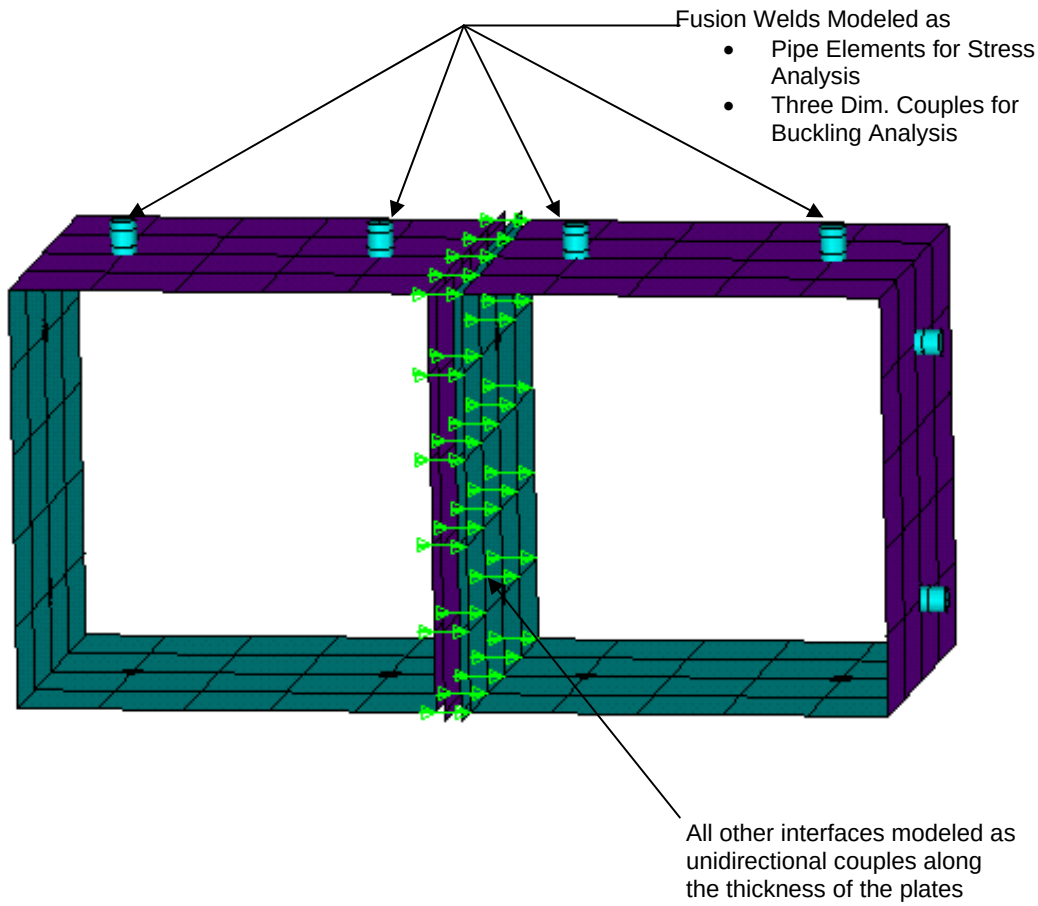


Figure 2.10.5-22
Fuel Basket Buckling Analysis Finite Element Model Node Couplings

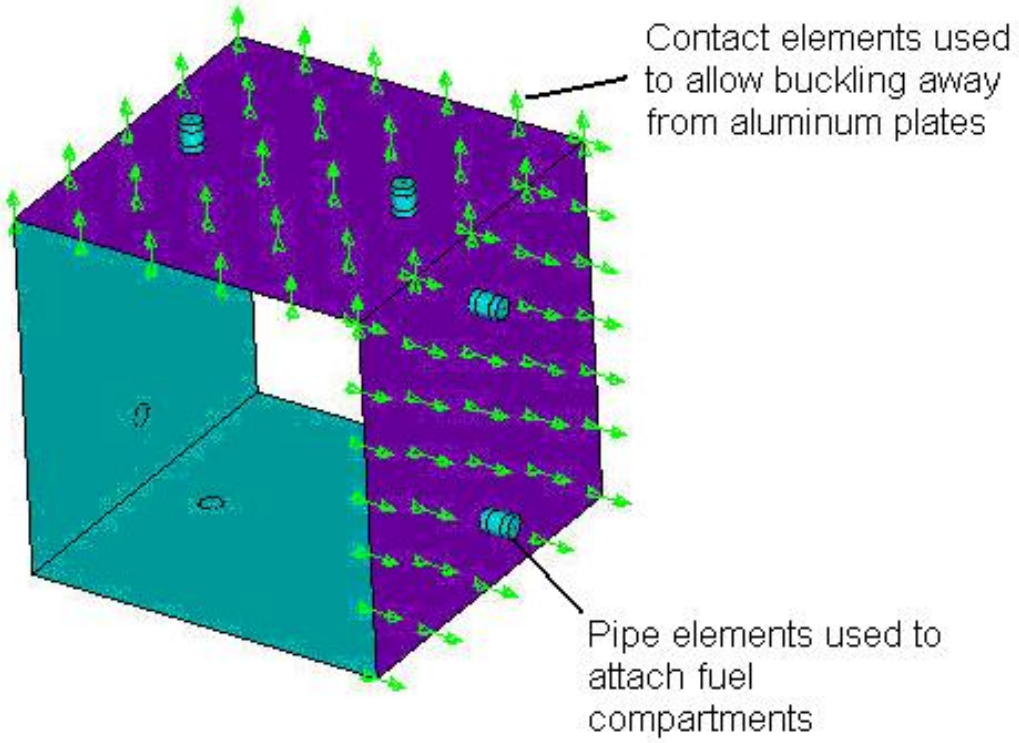


Figure 2.10.5-33
Fuel Compartment Interface Elements used in Sensitivity Study

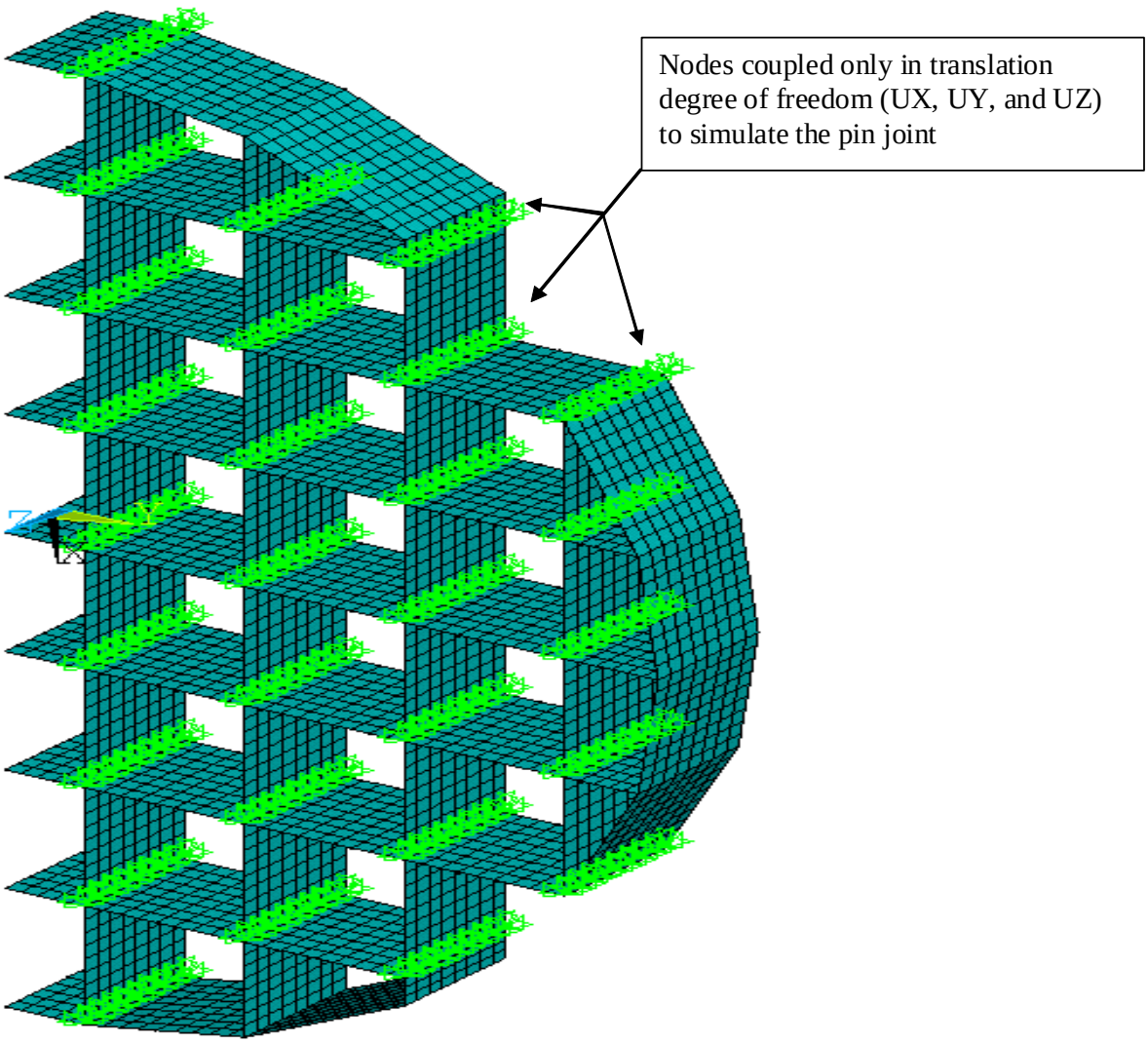


Figure 2.10.5-37
Basket Finite Element Model to Calculate the Maximum Compressive Load

Table 2.10.9-1
Comparison of Calculated vs. Measured *g* Loads

30 foot Drop Orientation	***g* Load Measured by Drop Test (Appendix 2.10.9)**	***g* Load Calculated by ADOC (Appendix 2.10.8)**
90° End Drop	54*g* Axial [(1)]	49*g* Axial
0° Side Drop	51*g* Transverse [(1)]	51*g* Transverse
CG Over Corner	34*g* Axial	32*g* Axial
20° Slap Down *(Second Impact)*	*58g* [(1) (2)] (Transverse + Rotational) *(measured location closed to top end of basket section)*	*76g* [(3)] (Transverse + Rotational) *(reported at outer edge of the cask lid)*

Notes:

(1) The g-loads reported in the tables contained in Section 2.10.9.6 of this Appendix are based on peak value of the raw data recorded from the test results. For design purposes, the region around the peak response has been smoothened to remove the scatter of the test data and provide a representative maximum acceleration of the test data. The resulting maximum acceleration values are reported in this table.

(2) The top/bottom ends of the cask body are enclosed by the impact limiters. There are no accelerometers located either on the impact limiters or on the cask body inside of the impact limiters. As shown on Figure 2.10.9-2, the accelerometers are located next to the impact limiters. This location is coincident to the top end of the basket section. The g load measured at this location represent the maximum combined transverse and rotational g load for the basket structural analysis due to the slap down drop.

(3) The maximum combined g load is 76g (27g + 49g) at the top end of the cask body (at the outer edge of the cask lid). For the maximum combined g load at locations other than the top end of the cask body (such as at the top end of the basket section as measured from the drop test), the rotational g load needed to be ratioed by the distance to the package CG and then added the transverse g load (27g).

Table 2.10.9-2

(deleted)

3.5.1 Fire Accident Evaluation

The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-40. This is assured as long as the metallic seals in the lid remain below 536°F and the cask cavity pressure is less than 100 psig. A full-length, 90 degree symmetric cask model as described in Section 3.4.1 is used for the evaluation. The model is modified to represent two crushed impact limiters as described in Section 3.5.3

Reference [8] reports an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-^{o}F for a railroad tank car fire test. The same parameter is utilized for the HAC fire accident evaluation.

3.5.2 Boundary conditions for the HAC

The boundary conditions described in Section 3.4.1. are modified for the HAC fire. During the pre-fire and post-fire phases, convection and radiation from the external surface of the model replicate the NCT analysis (100° F ambient). During the fire phase, a constant convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F is used. All gaps are removed during the fire and restored immediately after the fire. This assumption is conservative in that it ensures maximum heat transfer into the cask during the fire and minimum heat transfer from the cask during the post-fire cooling period. As required by 10CFR71.73 [1], a 30 minute 1,475 °F temperature fire with an emittance of 0.9 and a surface absorptivity of 0.8 is applied to the model. An emissivity of 0.9 and an absorptivity of unity are used for the cask external surfaces after the fire accident condition in order to bound the problem.

The sensitivity study that documents the effects of fire emissivity of 1.0 on thermal performance of the TN-40 cask is discussed in Appendix 3.7.3.

A detailed description of the model including the method used to calculate the maximum fuel cladding temperature and the average cavity gas temperature is provided in Section 3.4.1. The decay heat load used in this analysis corresponds to a conservative total heat load of 22 kW from 40 assemblies (0.55 kW/assy) with a peaking factor of 1.2 even though the design basis total heat load for transportation condition is 21 kW per cask.

3.5.3 Crushed Impact Limiter Models

In order to maximize the effect of the fire on cask components during and after the fire accident, the impact limiter finite element model developed in Section 3.4.1.2 is modified to reflect deformation due to a 30 foot drop. The maximum amount of crush experienced by the impact limiter in a given direction is assumed to occur everywhere on the limiter. Crushed impact limiter configurations based on side, corner and slap down drops are considered:

1. A crushed impact limiter corresponding to the side drop resulting in the shortest radial distance between the fire ambient and the cask surface. The maximum radial deformation of top and bottom impact limiters is 13.42 in. and 13.58 in. respectively. The impact limiters are thus modeled with a uniform

3.7.3 SENSITIVITY STUDY FOR EFFECTS OF FIRE EMISSIVITY

3.7.3.1 Discussion

A fire emissivity of 0.9 was used in Section 3.5.2 to calculate the fire radiation heat transfer to the TN-40 cask during the fire. Assuming conservatively, the fire as a black body, an emissivity of 1.0 can be used for the fire. The effect of this assumption is determined for the TN-40 cask in a sensitivity analysis in this section. According to Section 3.5.4, the case of a deformed impact limiter with a torn middle segment represents the bounding accident case and is chosen for the sensitivity analysis. The methodology and assumptions used in the sensitivity analysis are the same as those described in Section 3.5 for the HAC thermal evaluation except for the increase of the fire emissivity from 0.9 to 1.0.

A comparison of the maximum TN-40 component temperatures based on fire emissivities of 0.9 and 1.0 is shown in Table 3.7.3-1.

As seen from Table 3.7.3-1, the effect of increasing the fire emissivity from 0.9 to 1.0 on maximum fuel cladding temperature is an increase of 2 °F. The predicted maximum fuel cladding temperature of 531 °F (277 °C) is well within the allowable fuel cladding temperature limit of 1058 °F (570 °C) [1], [2] for accident conditions.

The largest effect of increasing the fire emissivity from 0.9 to 1.0 occurs at the cask outer shell when it is directly exposed to the fire. The other components remain shielded from the fire so that the inner shell temperature increases by only 17 °F and the cask rail temperature increases by only 9 °F.

These temperature increases are relatively small and last for a short period of time and therefore do not affect the thermal and structural performance of the TN40 transport cask.

The containment seals are protected from direct fire exposure by the impact limiters. The effect of increasing the fire emissivity from 0.9 to 1.0 on the maximum seal temperatures is limited to 8 °F for a short period of time after the fire. The transient and the steady state temperatures of the containment seals remain well below the temperature limit of 536 °F (280 °C) [3]. Therefore the containment function of the seals is unaffected by the increase of the fire emissivity from 0.9 to 1.0.

The time-temperature histories of the maximum component temperatures for the cask outer shell and fuel cladding from the sensitivity study with fire emissivity of 1.0 compared to those from the original model with fire emissivity of 0.9 are shown in Figure 3.7.3-1.

As seen in Figure 3.7.3-1, the TN-40 cask component temperatures decrease through the cool-down period. The small differences seen in Table 3.7.3-1 between the steady state temperatures are caused by the fact that the transient temperatures at 40 hours after fire are picked to bound the steady state temperatures.

3.7.3.2 Conclusion

The effect of increasing the fire emissivity from 0.9 to 1.0 lasts only for a short period of time on the outermost components of the cask exposed to fire. The function of the other cask component remains unaffected by this change in the fire emissivity.

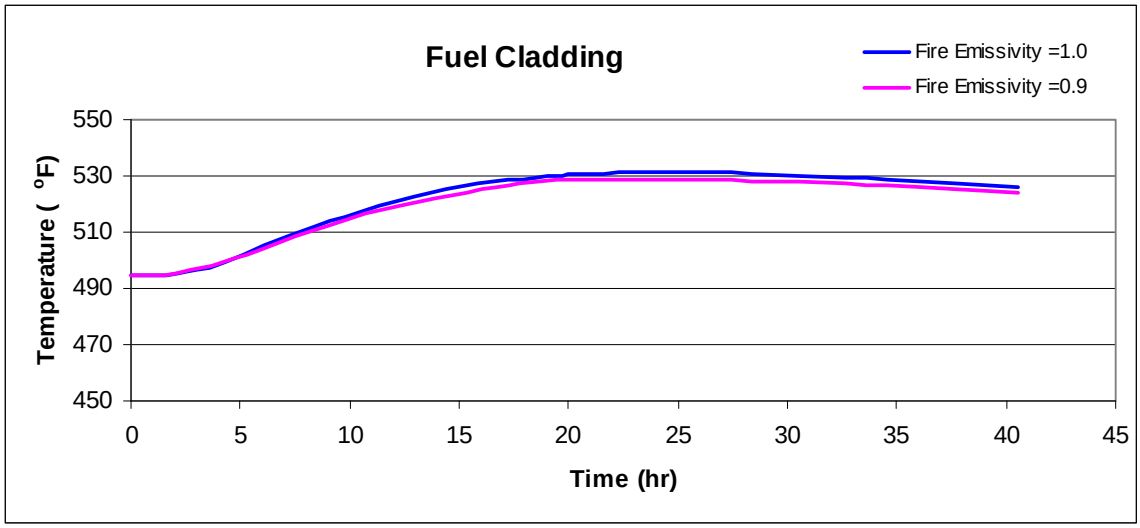
3.7.3.3 References

1. *Code of Federal Regulations, 10CFR71, "Packaging and Transportation of Radioactive Materials."*
2. *USNRC, SFPO, "Cladding Consideration for the Transportation and Storage of Spent Fuel," Interim Staff Guidance ISG-11, Rev. 3.*
3. *Helicoflex High Performance Sealing Catalog, Carbone Lorraine, Helicoflex Components Division, ET 507 E 5930.*

***Table* 3.7.3-1**
Maximum Component Temperature for 22kW Heat Load for Fire Emissivity =1.0 and 0.9

	Transient	***Temperature (F)***		***Steady State Temperature*** [1]			
	ε_f=1.0	ε_f=0.9		ε_f=1.0	ε_f=0.9		***Allowable Limit (°F)***
Component	T_{max} (°F)	T_{max} (°F)	ΔT_{max} (°F)	T_{max} (°F)	T_{max} (°F)	ΔT_{max} (°F)	
Fuel cladding	*531 (26 hours after fire)*	*529 (26 hours after fire)*	*+2*	*526*	*524*	*+2*	*1058 [2]*
Basket (fuel compartment)	*482 (20 hours after fire)*	*480 (20 hours after fire)*	*+2*	*476*	*474*	*+2*	*(2)*
Cask rail/shim	*339 (0.9 hours after fire)*	*330 (1.0 hour after fire)*	*+9*	*284*	*283*	*+1*	*(2)*
Cask inner shell[4]	*420 (end of fire)*	*403 (end of fire)*	*+17*	*278*	*277*	*+1*	*(2)*
Gamma shield shell	*726 (end of fire)*	*694 (end of fire)*	*+32*	*274*	*273*	*+1*	*(2)*
Lid O-ring seal[3]	*333 (1.7 hours after fire)*	*325 (1.0 hour after fire)*	*+8*	*230*	*229*	*+1*	*536 [3]*
Cask outer shell	*1138 (end of fire)*	*1084 (end of fire)*	*+54*	*253*	*252*	*+1*	*(2)*
Impact limiter surface	*1478 (end of fire)*	*1431 (end of fire)*	*+47*	*147*	*147*	*0*	*(2)*
Average cavity gas	*389 (13.9 hours after fire)*	*386 (10.2 hours after fire)*	*+3*	*375*	*374*	*+1*	*-*

[1] Thermal analysis results at 40 hours after the end of fire is used conservatively to bound the steady state temperatures. Actual steady state temperatures in both runs will achieve the same values.
[2] The components perform their intended safety function within the operating range.
[3] The elements between the cask shell flange and cask lid at radius (cylindrical x-coordinate) between 36.43" and 41.38" and height (cylindrical z-coordinate) between 164.55" and 171.55" represent the location of the lid seal in the model.
[4] Includes shell flange.



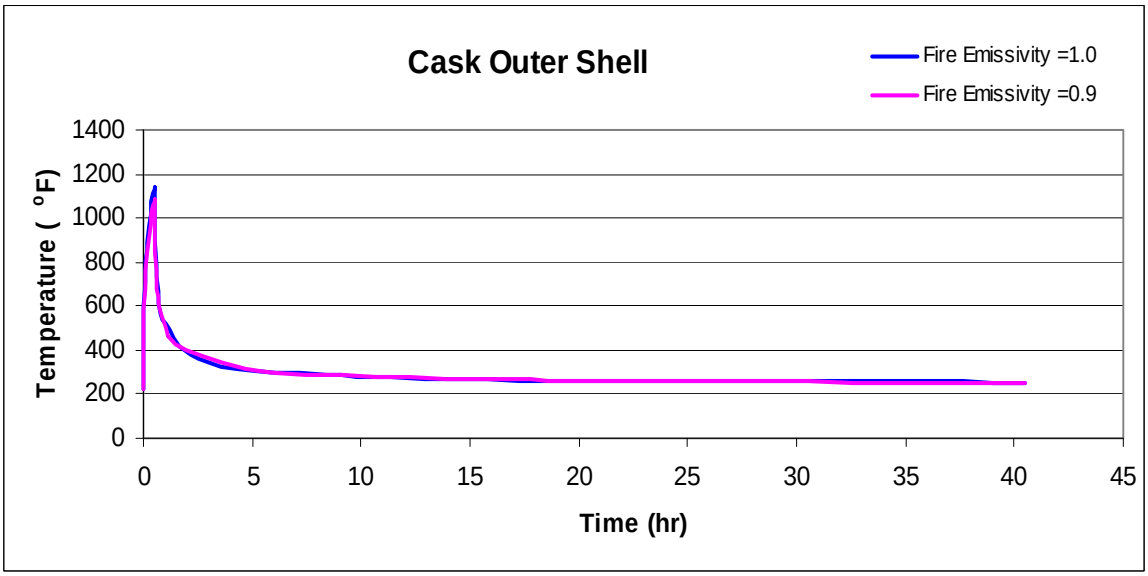


***Figure* 3.7.3-1**
Results of the Sensitivity Study

The third source is from the fuel itself. A breach in the fuel cladding may allow radionuclides to be released from the fuel to the interior of the cask. There are three types of radionuclides releases associated with the breaches in the fuel rod cladding: gaseous, volatiles and fuel fines.

For conservatism, it is assumed that crud spallation and cladding breaches occur instantaneously after fuel loading and closure operations. Therefore, all radioactivity is readily available for release if a leak occurs.

The containment analysis is based on the void volume within the TN-40 cask.

The cavity volume is 6.64E+05 in^3 (volume of a 72 in. x 163 in. cylinder). From SAR Table 2-6, the basket contains 6,610 lb of steel and 8,080 in^3 lb of aluminum. Using a density of 0.29 lb/in^3 for steel and 0.098 lb/in^3 for aluminum, the basket volume is calculated as 1.05E+05 in^3. With a fuel rod OD of 0.426 in. and a rod length of 152 in., the volume for 40 fuel assemblies each containing 179 fuel rods is calculated to be 1.55 E+05 in^3. The volume for the fuel assembly hardware (spacer grids, guide tubes, and instrument tube) is approximately 218 in^3 per assembly. Thus the calculated fuel assembly volume (40 assemblies) is 1.64E+05 in^3. Calculating the cask void volume:

$$\begin{aligned}\text{Cask Void Volume} &= 6.64\text{E+05 in.}^3 - 1.05\text{E+05 in.}^3 - 1.64\text{E+05 in.}^3 \\ &= 3.95\text{E+05 in.}^3 \\ &= 6.46\text{E+06 cm}^3\end{aligned}$$

<u>Source Activity from the Fuel</u>

The fuel transported in the TN-40 transport packaging has a maximum assembly average initial enrichment of 3.85 wt% U-235, 45,000 MWD/MTU bundle average exposure and a minimum of 15 - 25 year cooling time provided the fuel acceptance criteria of Section 1.2.3 have been met. As discussed in Chapter 5, Section 5.2, numerous SAS2H [8] evaluations were performed to determine the design basis fuel for shielding. These SAS2H analyses were also evaluated to determine the bounding fuel parameters for the containment analysis. The bounding SAS2H evaluation was performed for the 14 x 14 Westinghouse standard fuel assembly with 39,000 MWD/MTU burnup, enrichment of 3.3 wt. % U-235 and a cooling time of 15 years. It is assumed that 40 design basis fuel assemblies are loaded in the TN-40 cask transport packaging. The radionuclide inventory consists of activity from iodine, fission products that contribute greater than 0.1% of the design basis fuel activity and actinides that contribute greater than 0.01% of the design basis activity. Tritium is also included although it contributes slightly less than 0.1% of the design basis activity. The radionuclide inventory is presented in Table 4-1.

Source Activity from Release of Volatiles

The source activity concentration inside the TN-40 due to the release of volatiles, $C_{volatiles}$, is calculated using the following equation [4].

5.0 SHIELDING EVALUATION

5.1 Discussion And Results

Shielding for the TN-40 package is provided mainly by the cask body. The cask body is made up of the containment vessel, the gamma shielding and the lid. For the neutron shielding, a borated polyester resin compound surrounds the gamma shield shell radially. Additional shielding is provided by the steel outer shell surrounding the resin layer and by the steel and aluminum structure of the fuel basket.

For transport, wood filled impact limiters are installed on either end of the cask and provide additional shielding for the ends and some radial shielding for the areas at either end of the radial neutron shield. Figure 5-1 shows the configuration of the package shielding. Table 5-1 lists the compositions of the shielding materials.

The fuel assemblies acceptable for transport in the TN-40 are listed in Section 1.2.3. Using the SAS2H/ORIGEN-S modules of SCALE [1], source terms are calculated. The bounding design basis fuel for dose rate has an initial enrichment of 2.35 wt% and a total maximum bundle-average burnup of 42,000 MWD/MTU with a 24.4 year decay time.

The Westinghouse 14x14 standard fuel assembly contains the maximum heavy metal weight (Section 1.2.3) which results in bounding neutron and gamma source terms and is therefore identified as the most conservative fuel assembly. Section 5.2 describes the source specification and Section 5.4 describes the shielding analysis performed for the TN-40 cask. The shielding analysis models are described in Section 5.3.

Normal Conditions of Transport (NCT) are modeled with the neutron shielding and impact limiters on TN-40 intact. This shielding calculation is performed using the Monte Carlo computer code MCNP [5,*9*]. Dose rates on the side, top and bottom of the TN-40 package are calculated for the various sources described in Section 5.2 and summed to give a total gamma and neutron dose rate.

Hypothetical Accident Conditions (HAC) assume that the neutron shield and the impact limiters are removed. This evaluation bounds the accident conditions since it is shown in Chapter 2 and Chapter 3 that the neutron shielding may be lost but the impact limiters remain on the cask during HAC. Shielding calculations for the HAC are also performed using MCNP.

The expected maximum dose rates (for NCT and HAC) from the TN-40 package are provided in Table 5-2. Although this dose rate evaluation is performed using design basis fuel, evaluations were performed to determine that 15 year minimum cooled fuel is also acceptable for certain burnup and enrichment combinations. These evaluations were performed to determine the fuel assembly parameters of burnup, percent initial enrichment and cooling time that would result in decay heat and radiological sources that would meet the decay heat requirements (Chapter 3), source terms for containment (Chapter 4) and radiological sources that provide dose rates less than the current

design basis fuel mentioned above and thus would be acceptable for transport in the TN-40 package. Section 5.2 describes these evaluations in more detail.

The shielding calculations considered effects of tolerances. Since dose rates along side of the transportation package are controlling, cumulative effect of tolerances (+.05/-.01" on 1.50" thk. inner shell and +/-.12" on 8.00" thk. gamma shell) of steel thicknesses and tolerances (+/-.12" on 4.50") of resin on side of the cask is considered. Only tolerances in thicknesses of the neutron shielding, cask inner and gamma shells have the profound effect on dose rates along side of the cask. Note, dose rates presented in Table 5-2 *and* Table 5-18 *account for the described tolerances. The effect of the tolerances on dose rates at various distances from ends and at radial distances from side other than 2 meters is not significant to the extent that the dose rates at the lower tolerances limits would exceed or become close to regulatory limits.*

.The following considerations should be accounted for when using dose rates in Table 5-2 *and* Table 5-18*.*

- *Design basis Westinghouse 14x14 Standard fuel assemblies with the bounding neutron and gamma source terms are utilized in the shielding evaluation*
- *The fuel qualification methodology calls for conservatively adjusting the enrichment / burn-up and cooling time of the loaded fuel assemblies (Table 5-8).*
- *There is a number of conservative simplifications utilized in radiological source term and MCNP calculation model. They are outlined in Section* 5.5*. Each simplification, in average, adds a few percentage points of conservatism to estimated dose rates but their cumulative effect is noticeable.*
- *Calculated dose rates are generally higher than measured dose rates demonstrating the conservatisms in the shielding analysis methodology.*

5.2 Source Specification

There are five principal sources of radiation associated with transport of spent nuclear fuel that are of concern for radiation protection:

- Primary gamma radiation from spent fuel,
- Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission),
- Gamma radiation from activated fuel structural materials and fuel inserts,
- Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask, and
- Neutrons produced by sub-critical fission in fuel.

The TN-40 package is designed to transport Westinghouse 14 x 14 class PWR fuel assemblies. The fuel assemblies acceptable for transport in the TN-40 are described in Section 1.2.3. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of guide

/instrument tubes. These parameters are the significant contributors to the SAS2H/ORIGEN-S model. The largest uranium loading results in the largest source term at the design basis enrichment and burn-up, thus the Westinghouse 14 x 14 standard is the bounding assembly type.

Table 5-3 provides characteristics of the design basis fuel assembly used in the source term analyses. The SAS2H/ORIGEN-S modules of the SCALE code are used to generate gamma and neutron source terms for the bounding Westinghouse 14 x 14 standard assembly. Source terms were generated for initial enrichments ranging from 2.00 wt% to 3.85 wt% U235 and the fuel is irradiated for a constant time of 400 effective full power days per cycle. Burnup values range from 17 GWD/MTU to 45 GWD/MTU using a specific radiation power between about 15 and 25 MW/assembly. A conservative operating cycle history is utilized with a 30 day down time between cycles. Details of the analyses are given in Section 5.2.5.

The source terms are generated for the fuel assembly active fuel region, the plenum region, and the end fitting regions. The fuel assembly hardware materials and masses on a per assembly basis are listed in Table 5-4. Table 5-5 provides the material composition of fuel assembly hardware materials. Cobalt impurities are included in the SAS2H model.

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.2, respectively [4]. These factors are used to correct for the spatial and spectral changes of the neutron flux outside of the active fuel zone. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

5.2.1 Axial Source Distribution

PWR plant operations data for over twenty 14 x 14 fuel assemblies with approximately 36 to 42 GWD/MTU burnup are averaged into a typical profile, shown as maximum profile in Figure 5-2. Also shown in Figure 5-2 is the axial profile from Reference [3] for 38-42 GWD/MTU burnup fuel. The third profile shown in Figure 5-2 is a bounding profile *and* used in this analysis. *The bounding profile is also applicable for fuel with blankets at the ends of active zone. The use of bounding profile ensures that the most penalizing profile is utilized that accounts for blanketed versus un-blanketed fuel. The bounding profile is derived from an evaluation that includes fuel assemblies with blankets. Enrichment used in the shielding analysis is "average" and accounts for blankets. Burn-up used in shielding analysis is "average" and accounts for blankets.*

Using burn-up profile accounting for presence of blankets with low enriched or natural uranium increases dose rates on side of the package.

The conservative axial profile containing 12 axial zones is utilized in the shielding evaluation. The top and bottom 17% of the assembly are divided into two zones each and the middle 66% divided into 8 approximately equal zones. The peaking factors range from 0.700 at the bottom and top, to a maximum of 1.16 just below the middle.

5.2.5 Fuel Qualification

As stated previously, an evaluation was performed to determine the fuel assembly parameters of burnup, percent initial enrichment and cooling time that would result in dose rates and decay heats less than the design basis fuel mentioned above and thus would be acceptable for transport in the TN-40 cask.

These analyses were carried out using the SAS2H depletion module from the SCALE computer software and MCNP. For all SAS2H calculations the latest SCALE 44 group ENDF/B-V (44groupndf5) library was used. MCNP calculations used the default cross section libraries.

An MCNP model was utilized to calculate a response function at 2 meters from the transportation vehicle. Segmented surface tallies at 2 meters *radial distance from side of the transportation package* obtained from the MCNP analysis are essentially a source to dose conversion factor as a function of energy, (for gamma). These conversion factors are multiplied by an “adjustment” factor to account for the axial burn-up profile of the fuel to obtain the response functions that are used for the fuel qualification evaluation. The axial burn-up profile utilized in fuel qualification analysis is the same as the one utilized in the design basis dose rate shielding calculation. For neutrons, since a bounding energy spectrum is used, the response function calculated is just a total source to dose factor.

More than 200 SAS2H analyses were performed to determine gamma, neutron and thermal source terms as a function of burn-up, enrichment and cool time. The gamma source was obtained as a function of energy. *Note however that even the source encompasses 0.05 to 10 MeV energy range the primary gamma radiation dose rate is essentially contributed by the source energy groups within 0.8 to 3.0 MeV range. Response function entries were generated for each of the most contributing to the primary gamma radiation dose rate energy group. Two meter radial primary gamma dose rates were estimated/calculated by multiplying those entries and the appropriate SAS2H gamma energy source and performing a summation over the energy groups.* Since a bounding energy spectrum was used for the neutrons, the neutron and secondary gamma dose rates were calculated by multiplying the neutron response function and the total SAS2H neutron source and the secondary gamma response function and the neutron source. The estimated total 2 meter dose rate at approximately 20 cm above the middle of the active fuel was determined by combining the gamma and neutron dose rates. *Two meters radial distance is considered from side of 10’ wide transportation platform, which is 2 feet less than the width of impact limiters.* Provided the decay heat is less than 525 watts, the estimated dose rate must be less than 9.8 mrem/hr for the fuel to be qualified for transport.

Response function is used for ranking radiological sources by the dose rates they result in at two meters from surface of the package. Dose rates along side of the package are controlling. This is because dose rate at any radial distance from the side is larger than the dose rate at the same axial distance from ends of the cask. The maximum dose rate along side at 2 meters radial distance from the package side occurs at the axial

coordinate that is around the middle of the in-core region. Because neutron radiation source is associated with in-core region only and intensity of primary gamma radiation sources in energy groups that contribute the most to the primary gamma radiation dose rates is substantially greater there, the dose rate is essentially contributed by radiological source from in-core region.

Note that the total intensity of the primary gamma radiation (PGR) source in the in-core region is greater than in any of the end fittings regions by a factor of 1000 or higher. The spectrum of PGR source in the in-core region is distributed within a fairly broad energy range, 0.05 to 10.0 MeV, with a fraction of 0.04 in the 1.00 to 1.66 MeV range (~0.5 fraction in the 0.4 to 3.0 MeV range). Intensity of the PGR source in the end fittings of the FAs is mainly concentrated within 1.00 to 1.66 MeV range but its intensity is lower than the intensity in that energy range for the in-core region by a factor of 150 greater. This is true for burn-up, enrichment and cooling time (BECT) combinations (with some variations in numbers depending on a BECT) considered during qualification of fuel assemblies. Therefore, the contribution of radiological sources in the end fittings regions to the total dose rate at the controlling location near the cask side is nearly negligible in comparison with the in-core region. This implies that one can use any arbitrary source in the end fitting regions or no sources at all during ranking of radiological sources. This will only change the shape of the dose rate distribution on the cask ends (which, as pointed out earlier, are not controlling) and along the side of the cask near end fittings regions and interfaces between them and the in-core region, but it will not affect the dose rate distribution at axial coordinates over the middle of the in core region. Because of the negligible contributions of the end fittings regions to the dose rate at the location of interest, any deviations in material densities and elemental composition, thicknesses and layout of shielding materials near the cask ends would also, at most, have an effect on the shape of the dose rate distribution along the cask side near its ends but would not change the dose rate at the location of interest, at the axial coordinates near the middle of the in-core region. The response function is used only for determination of the radiological source that results in the highest dose rate at two meters radial distance from side of the cask. Therefore, there is no need to model end regions of the cask with great details, accuracy and consistency with the specifications in MCNP computational models when determining the response function. The only consistency is required for side of the cask and in-core-region. That is why it is said MCNP model for the response functions is essentially the same as the model for the design basis shielding calculation, where details, accuracy and consistency in modeling may have greater significance.

Response function (RF) entries are calculated for each energy bin, for each axial region with MCNP. One MCNP run calculates entries just for one energy group but for each region. However, such a calculation does not account for the fact that intensities of the sources in each axial region are different. Total intensity of the PGR source in the in-core region is greater than the source intensity of any of the end fittings regions by more than a factor of 1000. Because of this, "raw" response function data extracted directly from MCNP outputs and relevant to end fittings regions has to be scaled ("weighted") by the factor of 1e-3 prior to using them for ranking radiological sources.

Consider a single entry for in-core region corresponding to, say, 1.00 MeV to 1.33 MeV range of the PGR source. The physical meaning of the RF for that entry is that it is equal to the dose rate due to one source particle within this energy range in the in-core region of fuel assemblies inside of the cask at two meters radial distance from side of the cask. "N" such particles would result in N times larger dose rate. The same applies for the in-core region RF in other energy bins of the PGR spectrum. The same description applies for the RF entries due to other axial regions of the fuel assemblies inside of the cask. Therefore, knowledge of the radiological sources and RF for each axial region allows the calculation of the dose rates at the location of interest. To simplify the ranking of the sources from thousands of BECTs one can sum intensities of radiological sources for each energy group in all the axial regions and multiply them by a combined response function. Combined radiological sources can be easily calculated directly with a proper SAS2H\ORIGEN-S model. There is no need to calculate the sources for each axial region individually and sum them "manually" when ranking the sources. The combined radiological source (from the four regions of interest) and the combined response function greatly facilitate ranking of radiological sources. For this purpose, it is emphasized that spectral the structure is preserved in the combined radiological source and combined response function. That means the summation involves only entries within each individual energy bin. The entries of the combined response function are equal to the sum of the corresponding entries from each axial region. The summation takes into account the appropriate "weighting" of the entries from end fittings regions.

The calculated dose rate and decay heat along with the cooling time are then utilized according to the steps above to determine the bounding radiological source term. The final design basis radiological source term was generated by adding the TPA source term to the fuel/hardware source term. The cooling times calculated are reduced to a simplified look up table as a function of spent fuel parameters to summarize the loading parameters for the TN-40 transport package and are shown in Table 5-8.

Table 5-9 shows the results of the evaluation which define the spent fuel assembly cooling times to meet radiological and decay heat limits necessary for burnups ranging from 17 GWD to 45 GWD and enrichments between 2.0 wt% and 3.85 wt%. The TN-40 package containing fuel assemblies with parameters defined in this table will meet the dose rate and thermal criteria for transport. Table 5-10 shows the estimated dose rates and Table 5-11 shows the calculated decay heat corresponding to the cooling times shown in Table 5-9. All assemblies producing a decay heat of less than 21 kW per package or 525 watts per assembly are radiation (dose rate) limited. A fuel qualification table (FQT) for loading purposes based on this evaluation is provided in Table 1-2 (also shown in Table 5-8). The FQT is generated by conservatively rounding the cooling times shown in Table 5-9 up to the nearest value greater than 15 years.

5.3 Model Specification

The Monte Carlo computer code MCNP [5] is used for calculating the gamma and neutron doses in this analysis. *A more advanced version of the code MCNP [9] is also*

employed to determine the dose rates for the models that include tolerances as described in Section 5.4.

5.3.1 Description of Radial and Axial Shielding Configuration

Two base models were constructed. The first model corresponds to the neutron transport problem and the second is the gamma. Variance reduction was accomplished by means of importance zoning followed by weight windows. The importance function was created to balance the particles (per volume) throughout the problem geometry. The process used to do this was an iterative approach starting with basic attenuation factors for the shielding materials. The neutron importance function developed was also applied to the secondary gammas.

The test importance functions were then run in conjunction with the weight window generator. The weight windows calculated were inserted into the final MCNP runs. Weight windows were only used in the gamma cases.

The models were used to calculate both the axial and radial dose rates. The impact limiters and radial neutron shielding were removed for the HAC evaluation.

Sections 5.3.1.1 and 5.3.1.2 describe the shielding model (for the vicinity immediately around the cask) developed for the TN-40 under NCT and HAC.

5.3.1.1 NCT Radial and Axial Shielding Configuration

One shielding configuration is used for the TN-40 NCT design. The model is a complete three dimensional simulation of the TN-40 transportation package. The 72 inch diameter interior cavity of the cask is modeled with a discrete representation of the basket and fuel structure. Each fuel assembly is divided into four axial zones. The bottom zone represents the lower end fittings, the middle zone the active fuel region and the upper zones represent the plenum and upper end fittings, respectively. The axial locations of the plenum and the end fittings of the fuel assembly are similar to those provided in Reference [8]. The modeled active fuel length is 144 inches and the plenum length is 7.14 inches. The modeled bottom end fitting and top end fitting lengths are 3.08 inches and 3.5 inches, respectively. The fuel, end fittings and plenum are homogenized within the each assembly envelope and the axial length of their respective zones.

The basket structure is modeled as a 0.755 inch thick grid of aluminum and steel panels. The periphery of the basket is modeled by several peripheral basket universes to best represent the geometry. The TN-40 package model is illustrated in Figures 5-3 through 5-6.

The impact limiters are modeled as wood surrounded by a 0.25 in. thick steel shell. The interior steel gussets are conservatively neglected. The wood is modeled mostly as redwood except two areas, as shown in drawings 10421-71-41 and -42 (Appendix 1.4.1), which are modeled as balsa. An aluminum spacer utilized under the top impact limiter is included in the model.

For the dose calculation around the TN-40, the source is divided into four separate regions: fuel, plenum, top end fitting, and bottom end fitting. The model is utilized in two separate computer runs consisting of contributions from the following sources:

- Primary gamma radiation from the active fuel and from activated hardware within the top end fitting, plenum region and bottom end fitting (axial and radial directions).
- Neutron radiation from the active fuel region and secondary gamma radiation from neutron interactions.

The sources in the active fuel region (gamma and neutron) are modeled as uniform radially but vary axially. The sources in the structural hardware regions (plenum, top end fitting, and bottom end fitting) are modeled as uniform both radially and axially. The results from the individual runs are summed to provide the total gamma, neutron and total dose for the package.

The statistical uncertainties are generally less than 5% for the majority of tallies except for local tally bins and the accident results. For the accident the neutron end dose rates have the highest relative error around 10%. The statistical uncertainties associated with the neutron dose rates on the top and bottom impact limiter surface are high, but since they contribute less than 1% (less than 0.1 mrem/hr) to the total dose this is acceptable.

The terminology for the dose locations is as follows. On the side of the cask results are reported on the surface of the cask (“contact”), at vertical planes extending up from a 10 feet wide rail car (“vertical planes”), at the diameter of the impact limiters to represent the top and bottom of the package (“top/bottom”), 1 meter from the steel cask body (1 meter accident) and 2 meters from the vertical planes.

The results indicate peaking near the top and bottom of the cask and streaming in the upper trunnion/above the neutron shield regions. These results are expected due to the reduced shielding in these areas. It was determined that the normal conditions peak external surface dose rate of *73* mrem/hr occurs just above the neutron shield. This is approximately a factor of *1.8* times higher than the average on the cask surface. The localized peaking at the top of the cask is due to the absence of the neutron shield at the top. Neutron streaming was observed through the trunnion itself. However, the total dose rates just outside the trunnion were nearly the same as those averaged around the entire circumference of the cask.

Table 5-2 presents the maximum calculated dose at contact, at the vehicle’s outer edge (assumed 10 ft wide vehicle), and at 2 m from the vehicle’s outer edge. The calculated total dose rates at the various *distances* around the package are *presented* in *Table 5-2, Table 5-18.*

For the HAC, Table 5-2 also presents the maximum calculated doses at 1 m from the cask body.

The dose rates for an individual at the end of the rail car are presented in Table 5-17. These results are presented as a function of the length of the rail car.

On average the dose rates are dominated by the neutron source term. The results indicate that typically the total dose rates are comprised of 33% primary gamma, 14% (n,γ) and 53% neutron. However, the primary gamma source produces the majority of the dose rate at the ends of the package; the average contribution from primary gamma is approximately 81%. This is a direct result of the neutron shielding from the wood in the impact limiters. As expected, the accident dose rates are produced mostly from the neutron (94%) source due to loss of neutron shielding material and impact limiter.

Typical average (*beyond* 130 cm above and below the active fuel midplane) contact dose rates on the side of the cask are approximately *41* mrem/hr (*~50*% neutron). At 2 meters from the side of the cask *impact limiters* the average dose rate is approximately *7.3* mrem/hr which is comprised of *3.7* mrem/hr neutron, *0.7* mrem/hr (n,γ) and *2.9* mrem/hr gamma. On the ends, the total contact dose rates are less than 7 mrem/hr with less than a 0.1 mrem/hr contribution from neutrons. *All these dose rates are at the lower tolerance limits of the shielding materials thickness on side of the cask. Addressed tolerances are specified at the end on Section* 5.1*.*

Axial distribution of the total dose rate at various radial distances from the side of the transportation package is presented in Table 5-18*. Note that the neutron shield extends from -187 cm to +205 cm axial range in the MCNP calculational model. The table shows that there is a dose rate increase at between +190 cm to +230 cm axial coordinate range when considering axial dose rates at radial distances not exceeding radius of impact limiters. Dose rate there is larger than at the middle of the cask because of the top trunnions, less steel shielding due to "flat area" near trunnions and no neutron shielding. Also, top and plenum region in MCNP calculation model include 13.0 years cooled gamma sources from BPRAs and TPAs. The bottom trunnions and the cask's side at axial coordinates less than -187 cm is encompassed by bottom impact limiter and there is no BPRA/TPA sources at the bottom region (the ratio of top/bottom gamma source strength is roughly a factor of 1.2). Because of that, such a behavior in the dose rate distribution near the bottom trunnions is less pronounced.*

Because of the lack of shielding (see Figure 5-3*) and radiological source concentration near the top trunnions, dose rates were examined at axial coordinates around the trunnions. Specifically, dose rates near the top trunnions, on the flat part around the top trunnions as pointed out with the "P1" callout on a sketch of* Figure 5-4*, are evaluated. Contact dose rate at this point is 61.3 mrem/hr (31.5 mrem/hr neutron, 4.8 mrem/hr (n,g) and 17.1 mrem/hr gamma). At 2 meters radial distance measured from side of impact limiters, total dose rate is 9.7 mrem/hr (5.0 mrem/hr neutron, 0.8 mrem/hr (n,g) and 3.9 mrem/hr gamma). Again, all these dose rates are at the lower tolerance limits of the shielding materials thickness on side of the cask. Addressed tolerances are specified at the end on Section* 5.1

The dose rate analysis was performed using MCNP [5*, 9*]. Selected inputs for MCNP are included in Section 5.6.

5.5 Uncertainties and Conservatism in Shielding Evaluation

Uncertainties manifest themselves in shielding analysis when determining both, radiological sources and dose rates.

Determination of bounding radiological sources and dose rates was carried out using the SAS2H depletion module from the SCALE and MCNP, NRC approved computer software, respectively. Systematic errors may be present in the result of an estimate based on a mathematical model or physical law. They can be either constant, or be related (e.g. proportional or a percentage) to the actual value of the measured\calculated quantity.

Such errors for isotopic predictions, decay heat estimates are quantified and presented in the SCALE manual.

Nature of such errors in MCNP simulations is discussed in Chapter 2, Volume I of MCNP 5 manual [0]. When interpreting results of Monte Carlo simulations with computer codes like MCNP one has to differentiate the systematic error, which is seldomly known, and accuracy. The difference between true value and estimated with the computational model is called the systematic error and refers to accuracy. Uncertainty in MCNP calculation refer to precision. It is quite possible to calculate highly precise results that are far from the physical truth because the nature of the shielding has not been modeled faithfully, for example, with substantial conservatism. Therefore, it is permissible to have dose rates estimates that are very close or even equal to exact regulatory limits if it is known that certain features of the shielding configuration, source terms calculation are conservatively modeled and the extent of their conservatism can be quantified.

There are a few sources of conservatism employed in the calculation of radiological sources and in the development of the shielding model. The sources of conservatism identified in the source terms calculation and MCNP shielding model are summarized below.

Uncertainties and Conservatism in Calculating Radiological Sources:

Gamma source terms for the in-core region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only.

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which encompasses two most prominent lines of ^{60}Co. As for fission products, the main contributors after six years of decay with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh (^{106}Ru), ^{137}Cs, ^{144}Pr (^{144}Ce), ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh (^{106}Ru), ^{147}Sm, ^{142}Ce and ^{144}Pr (^{144}Ce) are the strongest emitters at energies greater then 2.0 MeV. The accuracy of the gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate

than those for actinides because the calculation of their inventory has less uncertainty [1]. Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times greater than 10 years [1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of ^{244}Cm is calculated. Per SCALE 4.4 [1], reported experimental ^{244}Cm densities are accurate within $\pm$20%. The gamma emission intensity from ^{244}Cm, which is proportional to the quantity of ^{244}Cm in the actinide inventory, is bounded by this value.

Primary gamma radiation dose rates are mainly due to radiological source in 0.8 to 3.0 MeV energy groups. Intensity of gamma radiation source in those energy groups is essentially determined by the following isotopes: ^{60}Co, ^{106}Rh (^{106}Ru) and ^{144}Pr (^{144}Ce). SCALE manual suggests that uncertainty in the source strength in that range is in the vicinity of 10 to 15% [1].

At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of considered burn-up, enrichment and cooling time combinations neutron radiation source is essentially represented by ^{244}Cm. The neutron spectrum is, therefore, relatively constant and its uncertainty is defined by precision, within $\pm$20%, with which the inventory of ^{244}Cm is calculated by SCALE for the fuel parameters addressed.

Therefore, uncertainties in calculated dose rates due to systematic errors in determination of radiological sources is not greater than 20 %.

The objective of the source term calculation is to determine the limiting spent fuel parameters that meet both radiation limits and thermal limits. The division between thermal limited and radiation limited spent fuel parameters is identified. Subsequently, a design basis fuel is determined with sufficient decay to ensure the dose rate estimate with a response function at 2 meters from side of 10' wide transportation platform is the highest among considered burn-up, enrichment, cooling time (BECT) combinations <u>and</u> less than or equal to 10.0 mrem/hr. The design basis shielding analysis using detailed MCNP model of TN40 transport cask utilizing the design basis source ultimately verifies compliance of the dose rates with the regulatory restrictions.

Conservatism in source terms evaluation:

- *BPRA+TPA sources are conservatively assumed in all 40 fuel assemblies for transport (i.e., one TPA+BPRA source per assembly).*
- *A dose rate limit of about 9.8 mrem/hr at 2 meters radial distance from side of 10' wide rail car is used for determination of cooling times.*

- *After radiation limited BECTs are identified the cooling times were conservatively rounded up to whole year increments except for the 2.35 wt. %, 42 GWD/MTU case which was rounded to 24.5 years for the fuel qualification.*
- *The design basis source term for the TN40 transportation cask is due to FA with 0.410 MTU load, 42 GWD/MTU burnup, 2.35 wt. % enrichment, 24.4 years cooling.*

Additional sources of conservatism in source terms evaluation is

- *Utilized in calculations Co-60 content in steel and other FA hardware materials is from reference [10]. The reference gives a conservative content for steel. The extent of conservatism can range from 10 to 15%, depending on FA design, age (i.e., when it is manufactured).*
- *Assumption that a reactor is operating at a constant power throughout an entire duration of cycles in the SAS2H\ORIGEN-S models.*
- *Bounding MTU load, 410 kgU/FA (up to 10 % of conservatism)*

Conservatism and Accuracy in Dose Rate Calculations using Monte Carlo Methods:

Conservatism in MCNP shielding models.

- *It is shown in the calculations, the limiting dose rates are typically at the sides of the cask. The fuel is assumed to be homogenized within the assembly envelope. This homogenization is conservative with respect to the side dose rate estimates.*
- *The steel boxes in the fuel compartments are modeled as 13 gauge steel at 0.09" thick. The reference drawings give a typical plate thickness of 0.10".*
- *The cask rails are assumed to be homogenized into an annular shell of pure aluminum with a thickness of 0.2 inch and outer radius equal to the cavity radius. This is a conservative assumption as it results in a reduced amount of aluminum shielding (near 80% of total mass).*
- *The periphery steel fuel boxes are homogenized into an annular shell on the inside of the homogenized rails. A thickness of 0.026" is assumed as it accounts for approximately 80% of the steel mass.*
- *The steel material on the impact limiters is only modeled on the ends. This is a conservative assumption and will only affect the end dose rates.*
- *The neutron poison material is conservatively neglected.*

*All the dose rates are evaluated with F2 or F4 type tallies in MCNP simulations. Precision of the results is represented with relative errors. According to MCNP manual [10] calculated with F2 and F4 type tallies dose rates are generally considered reliable if relative errors are about 10 % or less. The relative error can be decreased by longer duration of computer runs to allow accumulation of more statistical events for obtaining more accurate estimates. However, studying of tally fluctuation charts and "**nps**" logs*

suggests that tallies' mean estimates reached their "plateau" values and improving a precession of the estimates would not affect dose rate results to the extent that they would exceed regulatory limits.

Net Effect of Conservatism from Source Terms and Dose Rate Calculations:

The contribution of the sources of conservatism in SAS2H\ORIGEN-S depletion and MCNP shielding models to the extent of conservatism ranges from not greater than a percentage point to 10 to 15 %. Even assuming that each source accounts only in average for 2% results in cumulative ~30% increase of the dose rates. As mentioned earlier, uncertainties in calculated dose rates due to systematic errors in determination of radiological sources is not greater than 20 %. Therefore, the contribution from the identified sources of conservatism overlaps uncertainties inherited from determination of radiological source. It allows utilizing dose rate estimates that are very close or even slightly (10%) higher than regulatory limits on condition that a note is provided. For example, consider dose rates along side of the cask at two meters radial distance measured from edge of 10' wide platform (this is about 2' less than the width of impact limiters). They can be obtained by scaling dose rates in the last column of Table 5-18 *by the factor of 1.086 (=(0.5*144"*2.54+200)/(0.5*120"*2.54+200), where 120" and 144" are outer radii of the vehicle's vertical plane's (10' wide rail car) and impact limiters, respectively). The maximum would be 10.3 mrem/hr. However, there is at least 10% of conservatism in the dose rate estimate. Therefore, strictly speaking, even axial dose rates at two meters radial distance measured from side of 10' wide rail car are below 10.0 mrem/hr regulatory limit.*

5.6 References

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3. NUREG/CR-6801 (ORNL/TM-2001/273), March 2003, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses."
4. Luksic, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal," PNL-6906, UC-85, June 1989.
5. MCNP4C2, "Monte Carlo N-Particle Transport Code System," Los Alamos National Laboratory, CCC 701, RSIC, June 2001.
6. "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants," ANSI/ANS-6.6.1-1977, American Nuclear Society, Illinois, 1977.
7. EPRI-NP-5128, "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses," prepared by Pacific Northwest Laboratory, Virginia Power Company and EG&G Idaho National Engineering Laboratory, April 1987.
8. DOE/ET/47912-3 Vol. III, "Domestic Light Water Reactor Fuel Design Evolution," Prepared for U. S. Department of Energy Savannah River Operations Office, September 1981.
9. *"MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, January 2006.*
10. *S.B. Ludwig and J.P. Renier, "Standard and Extended-Burn-up PWR nd BWR Reactor Models for the ORIGEN2 Computer Code", ORNL/TM-11018, Oak Ridge National Laboratory, December, 1989.*
11. *O.W. Herman, J.P. Renier, and V.V. Parks, "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data", NUREG-CR-5625, ORNL/CSD-130, Martin Marietta Energy Systems Inc., Oak Ridge Natl. Lab. (1994).*

Table 5-2
Summary of TN-40 Dose Rates

(Exclusive Use)

Normal Conditions Of Transport	**Package *Contact Dose Rate* mSv/h (mrem/h)**			***Closed* Vehicle *Surface* mSv/h (mrem/h)**		**2 Meters from *Closed* Vehicle *Surface* mSv/h (mrem/h)**		
Radiation	**Top**	**Side**	**Bottom**	***Top/Bottom*[1]**	***Side* [2]**	**Top**	**Side[2]**	**Bottom**
Gamma	0.069 (6.9)	0.35 (35)	0.059 (5.2)	0.16 (16)	0.23 (23)	-	0.047 (4.7)	-
Neutron	0.0004 (0.03)	0.38 (38)	0.0008 (0.08)	0.16 (16)	0.26 (26)	-	0.049 (4.9)	-
Total	0.069 (6.9)	0.73 (73)	0.060 (6.0)	0.32 (32)	0.49 (49)	<0.069 (<6.9)	0.097 (9.7)	<0.060 (<6.0)
Limit	10 (1000)	10 (1000)	10 (1000)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

[1] *Top/Bottom is the exterior cylindrical surface of the impact limiters.*
[2] *Side is at vertical planes extending up from a 10 feet wide rail car ("vertical planes").*
[3] *Scaled from 2 m dose rates calculated from trunnion flat surface by a conservative factor of (0.5 x 101" x 2.54 + 200 m) / (0.5 x 144" x 2.54 + 200 m) – 101" is the outer resin housing's diameter and 144" is the impact limiter's diameter.*

Hypothetical Accident Conditions[4]	**1 Meter from Package Surface mSv/h (mrem/h)**		
Radiation	**Top**	**Side[5]**	**Bottom**
Gamma	0.43 (43)	0.32 (32)	0.28 (28)
Neutron	0.68 (68)	5.34 (534)	1.45 (145)
Total	1.11 (111)	5.66 (566)	1.73 (173)
Limit	10 (1000)	10 (1000)	10 (1000)

[4] The neutron shield and the impact limiters are removed.
[5] *Does not account for tolerances on side of the cask described at the end of Section 5.1. The effect of tolerances is less than 10%. It is not significant to the extent that dose rates would exceed regulatory limits.*

Table 5-10
Dose Rates (mrem/hr) at* 2 Meters *from Side of 10' wide Transportation Platform
Estimated *with Response Function*

	Assembly Average Initial Enrichment (Wt. % U235)								
Maximum Assembly Average Burnup (GWD/MTU)	**2**	**2.25**	**2.35**	**2.75**	**3**	**3.25**	**3.4**	**3.6**	**3.85**
17	10	10							
18	10	10							
19	10	10							
20	10	10							
21	10	10							
22	10	10							
23	10	10							
24	10	10							
25	10	10							
26	10	10	10						
27	10	10	10		10	10			
28	10	10	10	10	10	10			
29			10	10	10	10	10		
30			10	10	10	10	10		
31			10	10	10	10	10		
32			10	10	10	10	10		
33			10	10	10	10	10	10	
34			10	10	10	10	10	10	10
35			10	10	10	10	10	10	10
36			10	10	10	10	10	10	10
37			10	10	10	10	10	10	10
38			10	10	10	10	10	10	10
39			10	10	10	10	10	10	10
40			10	10	10	10	10	10	10
41			10	10	10	10	10	10	9.4
42			10	10	10	10	10	9.4	9.1
43					10	10	9.6	9.2	8.7
44						10	9.4	8.9	8.5
45						9.6	9.2	8.7	8.2

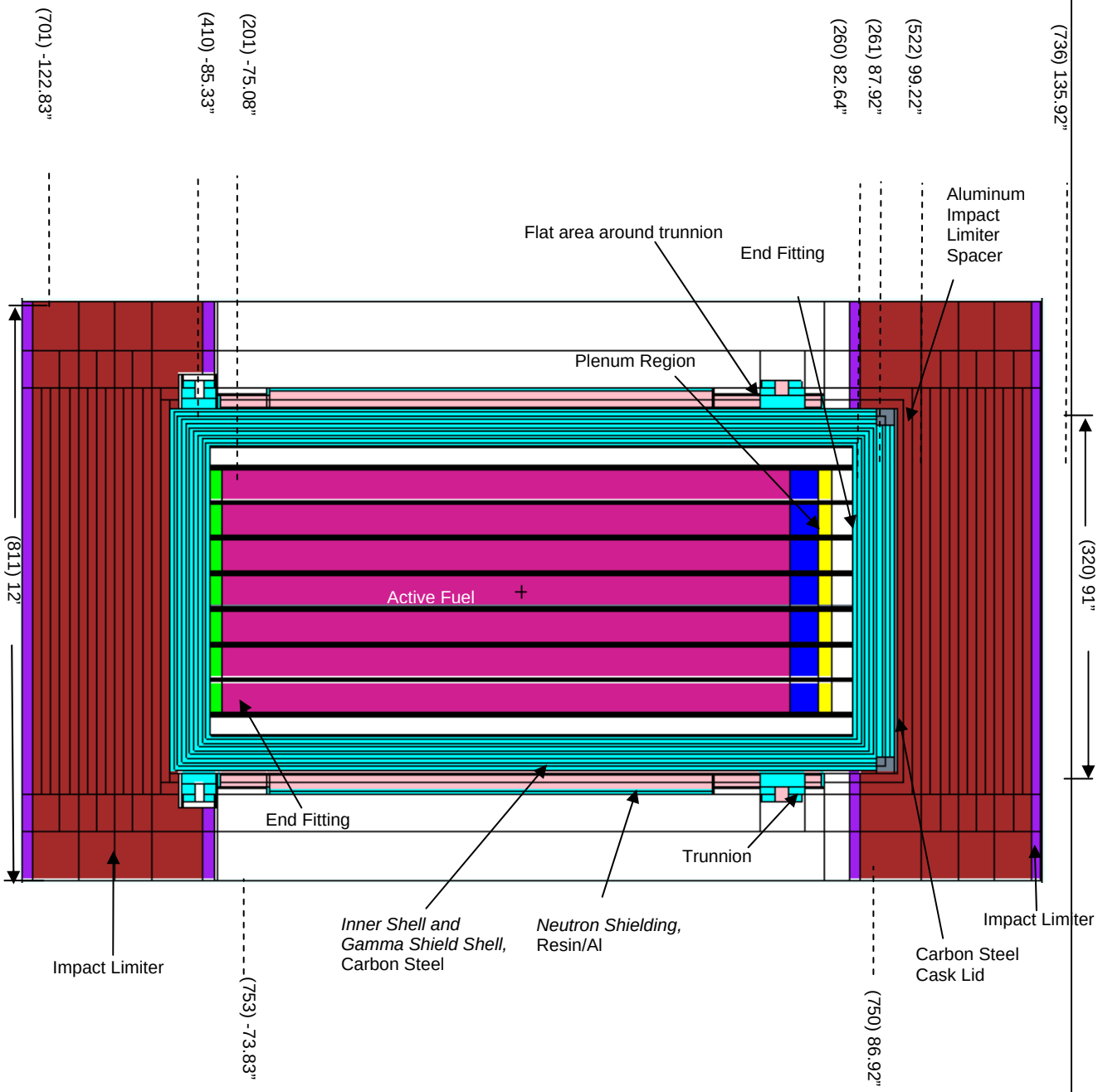
Table 5-18
Total Dose Rates Along Side of the Cask at Normal Conditions

Axial Range (cm). Note, ends of impact limiters are at approximately -312 and 346 cm.			***(1) Total Dose Rates (mrem/hr)***			
			(2)Contact	***Vertical Side***	***Bottom/Top***	***2 Meters(3)***
-306.9	*to*	*-190*	*19.7*	*1.36*	*1.90*	*1.7*
-190	*to*	*-170*	*48.0*	*28.4*	*16.7*	*5.7*
-170	*to*	*-150*	*26.2*	*23.5*	*19.0*	*6.5*
-150	*to*	*-130*	*31.0*	*24.4*	*20.1*	*7.1*
-130	*to*	*-110*	*36.8*	*27.4*	*21.6*	*7.6*
-110	*to*	*-90*	*41.3*	*30.2*	*23.4*	*8.1*
-90	*to*	*-70*	*44.0*	*32.2*	*24.7*	*8.6*
-70	*to*	*-50*	*45.5*	*33.2*	*25.9*	*8.9*
-50	*to*	*-30*	*46.5*	*34.2*	*26.4*	*9.2*
-30	*to*	*-10*	*47.3*	*34.0*	*26.8*	*9.3*
-10	*to*	*10*	*46.6*	*34.5*	*26.6*	*9.4*
10	*to*	*30*	*46.5*	*33.9*	*26.6*	*9.5*
30	*to*	*50*	*46.1*	*33.8*	*26.2*	*9.5*
50	*to*	*70*	*45.5*	*33.1*	*25.8*	*9.4*
70	*to*	*90*	*42.9*	*31.3*	*24.5*	*9.5*
90	*to*	*110*	*39.9*	*29.3*	*23.2*	*9.3*
110	*to*	*130*	*35.3*	*27.0*	*22.2*	*9.2*
130	*to*	*150*	*31.6*	*24.9*	*21.4*	*8.9*
150	*to*	*170*	*27.7*	*24.8*	*22.3*	*8.7*
170	*to*	*190*	*30.4*	*29.2*	*25.6*	*8.5*
190	*to*	*210*	*72.6*	*49.1*	*32.0*	*8.2*
210	*to*	*230*	*137*	*61.1*	*30.3*	*7.6*
230	*to*	*250*	*41.4*	*20.6*	*9.65*	*6.8*
250	*to*	*340.16*	*6.55*	*1.11*	*1.79*	*1.8*
Maximum Relative Error:			*1.9%*	*1.2%*	*1.8%*	*1.1%*

(1) Accounts for cumulative effect of tolerances (+.05/-.01" on 1.50" thk. inner shell and +/-.12" on 8.00" thk. gamma shell) of steel thicknesses and tolerances (+/-.12" on 4.50") of resin on side of the cask.

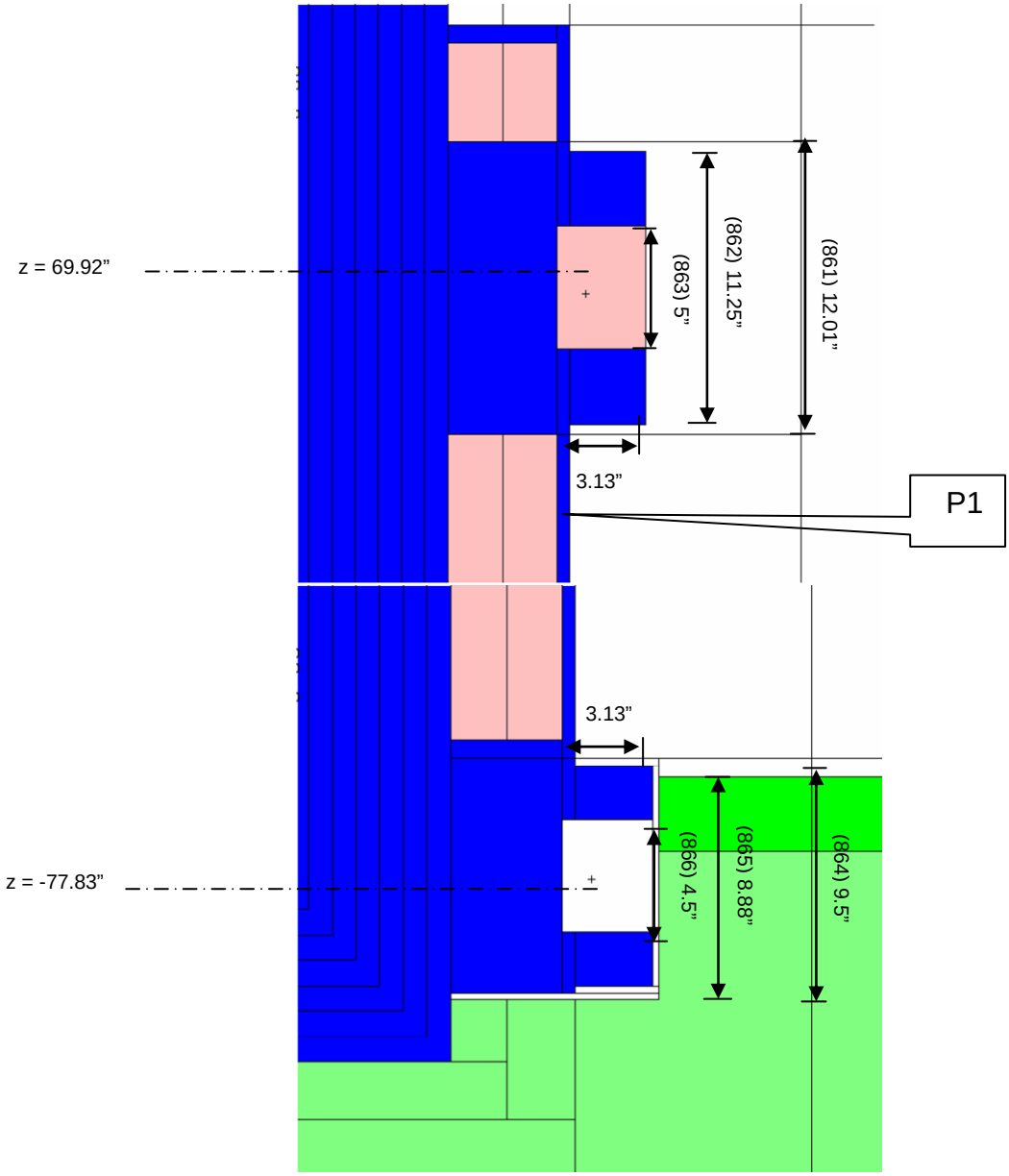
(2) Contact dose rates at axial coordinates less than -187 or greater than 220 cm. (in shaded cells of the table) are inside of impact limiters.

(3) This is at two meters radial distance from side of impact limiters. They are scaled from 2 m dose rates calculated from vertical plane on side (10' wide rail car) surface by a factor of (0.5x(120+2x10)"x2.54+200 m)/(2x144"x2.54+200 m); (120+2x10)" is the vehicle's vertical plane's and 144" is the impact limiter's diameter



Dimensions are shown in inches. Numbers in parenthesis correspond to surface numbers in MCNP model.

Figure 5-3
Side View of TN-40 Transport MCNP Model



Enlarged view of trunnion area. ***Dimensions are shown in inches. Numbers in parenthesis correspond to surface numbers in MCNP model.***

Figure 5-4
Detail Views of TN-40 Transport MCNP Model

7.0 OPERATING PROCEDURES

This chapter contains TN-40 transport package loading and unloading procedures that are intended to show the general approach to cask operational activities. A separate Operations Manual (OM) will be prepared for the TN-40 transport package to describe the operational steps in greater detail. The OM, along with the information in this chapter, will be used to prepare the site-specific procedures that will address the particular operational considerations related to the TN-40 cask. The operations required to convert the TN-40 cask from its storage configuration to its transport configuration are also described here.

7.1 Package Loading

For the TN-40 casks that have been loaded and used for storage under 10CFR72 requirements, use procedures for preparation of casks for transport described in Section 7.4.

7.1.1 Preparation for Loading

7.1.1.1 Once prior to placing the first fuel assembly into the cask. Verify each fuel assembly to be loaded satisfies the loading requirements listed in *Certificate of Compliance 71-9313*. This verification shall be performed by two independent individuals.

7.1.1.2 The assigned burnup loading value for each fuel assembly shall be obtained from a source controlled by the site's QA program and traceable to the TOTE or BURNUP output corresponding to when the fuel assembly was discharged from the reactor for the final time.

7.1.1.3 The assigned burnup loading value for each fuel assembly shall be from the TOTE or BURNUP computer codes. The value from these codes shall be reduced by a 1.04 factor to account for burnup uncertainties.

7.1.1.4 Once prior to inserting into cask, verify the identity of each fuel assembly. This verification shall be performed by two independent individuals.

7.1.1.5 Upon arrival of the empty packaging, on its transport vehicle (rail or heavy haul trailer) and shipping frame, perform a receipt inspection to check for any damages or irregularities. Verify that the records for the packaging are complete and accurate.

7.1.1.6 Remove the personnel barrier (if used), security device, the impact limiter attachment bolts, tie-rods, and the associated hardware, as necessary.

7.1.1.7 If they are mounted on the cask, remove the front and the rear impact limiters, as well as the top impact limiter spacer.

7.1.1.8 Remove the tie-down straps.

7.1.1.9 Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.

7.1.1.10 Using a spreader bar and lifting straps, lift the cask from the transport frame and lower it onto the upending/downending frame.

7.1.1.11 Attach the lift beam to the cask handling crane hook, and engage the lift beam to the two upper (top) trunnions.

7.1.1.12 Rotate the cask slowly from the horizontal to the vertical position.

7.1.1.13 Lift the cask and place it in the cask preparation area.

7.1.1.14 Disengage the lift beam from the cask.

7.1.1.15 Replace the neutron shield pressure relief valve with a plug.

7.1.1.16 Remove the lid bolts and the lid.

7.1.1.17 Remove the lid seal, vent and drain port cover seals and overpressure (OP) port seals and inspect the sealing surfaces. Install new seals in the vent and drain port covers and the lid. This step may be performed at any time prior to closing the loaded cask.

7.1.1.18 Visually inspect the bolts and the bolt hole threads for the lid, vent, drain, and OP ports.

7.1.1.19 Verify that the basket is installed in the cask. Verify that there is no foreign material in the cask.

7.1.1.20 Move the cask to the cask loading area using the lift beam attached to the top trunnions.

7.1.2 Loading

Note: The term “cask loading pool” is used to describe the area where the cask is to be loaded.

7.1.2.1 Lower the cask into the cask loading pool and fill the interior with water.

7.1.2.2 Disengage the lift beam and move it aside.

7.2.1.8 Using a spreader bar and lift slings, lift the cask from the transport vehicle and place it on the upending/downending frame.

7.2.1.9 Attach the lift beam to the cask handling crane hook, and then engage the lift beam to the two upper (top) trunnions.

7.2.1.10 Rotate the cask slowly from the horizontal to the vertical position.

7.2.1.11 Lift the cask from the upending/downending frame, and place it in the designated work area.

7.2.1.12 Disengage the lift beam from the cask, and move the crane as well as the lift beam from the area.

7.2.1.13 Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.

7.2.1.14 Remove the neutron shield pressure relief valve, and install the plug in the neutron shield vent hole.

7.2.2 Preparation for Unloading

7.2.2.1 Remove the vent cover.

7.2.2.2 Collect a cavity gas sample through the vent port quick-disconnect coupling.

7.2.2.3 Analyze the gas sample for radioactive material, and add necessary precautions based on the cavity gas sample results.

Note: If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed, and approved by the appropriate site personnel, as well as implemented to minimize worker exposures and radiological releases to the environment. These additional measures may include provision of filters, as well as respiratory protection and other methods to control releases and exposures to ALARA.

7.2.2.4 In accordance with the site requirements, vent the cavity gas through the vent port until atmospheric pressure is reached.

7.2.2.5 Remove the vent port quick-disconnect and the drain port cover. Attach the vent port adapter and the drain port quick-disconnect, if utilized.

7.2.3.8 Engage the lift beam on the upper (top) trunnions, and lift the cask out of the pool.

7.2.3.9 Using the drain port in the lid, drain the water from the cask in accordance with the procedures. This may be done either before or after lifting the cask out of the pool. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean demineralized water to facilitate decontamination. If the pool contains borated water, the effects of adding non-borated water to the pool must be considered.

7.2.3.10 Disconnect the drain line.

7.2.3.11 Move the cask to the decontamination area, and disengage the lift beam.

7.3 Preparation Of Empty Package For Transport

7.3.1 *Verify that the cask is empty and d*econtaminate the cask until acceptable *inner and outer* surface contamination levels are obtained *in accordance with DOT regulations for empty packages as directed in 49CFR173.428*.

7.3.2 Lubricate and install the lid bolts and torque them to 400 ft-lb. Follow the torquing sequence shown in Figure 7-1. A circular pattern of torquing may be used afterwards to eliminate further bolt movement.

7.3.3 Remove the plug from the neutron shield vent, and reinstall the pressure relief valve, making sure that it is operable and set.

7.3.4 If required by user or shipper, evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture.

7.3.5 Isolate the vacuum pump, and backfill the cask cavity with nitrogen.

7.3.6 Install the vent and drain port covers.

7.3.7 Re-engage the lift beam to the upper (top) trunnions of the cask.

7.3.8 Move the transport vehicle with transport frame installed into the loading position and place the upending/downending frame near the transport vehicle.

7.3.9 Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the upending/downending frame.

7.3.10 Rotate the cask from the vertical to the horizontal position.

7.3.11 Using a spreader bar and lift slings, lift the cask from the upending/downending frame and place it on the transport frame.

7.3.12 Install the tie-down straps.

7.3.13 Check if the surface dose rates and the surface contamination levels are within the regulatory limits for an empty cask *as given in 49CFR173.428*.

Note: If the impact limiters are going to be shipped separately, skip the next 4 steps.

7.3.14 Install the top impact limiter spacer on the front end of the cask. Then remove the spacer lifting eye bolts.

7.3.15 Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Loctite N-5000 or an equivalent, and torque to 60 - 80 ft-lb.

7.3.16 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.3.17 Render the impact limiter lifting lugs inoperable, by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.3.18 Perform a final radiation and contamination survey to assure compliance with *49CFR173.428*.

7.3.19 Install the personnel barrier.

7.3.20 *Remove or cover previous* DOT labels and placards *and attach an "Empty" label to the package* in accordance with 49 CFR 172*.450*, and prepare the final shipping documentation.

7.3.21 Release the empty cask for shipment.

7.4 <u>Other Procedures</u>

7.4.1 Preparation of Cask Used in Storage for Transport

The TN-40 cask is designed for storage as well as transport. The following steps are required to convert the TN-40 from its storage configuration to the transport configuration. In some cases, the casks which have been used for storage may not have the transport regulatory plate or nameplate installed on them. These plates must be installed prior to transport. In addition, some casks that have been used for storage may not have the impact limiter bracket mounts installed. As required, the mounts must be welded to the

outer shell for transport. The accessible surfaces of the cask shall be visually inspected for evidence of cracks in the carbon steel shell.

7.4.1.1 Verify each fuel assembly in the cask satisfies the loading requirements listed in *Certificate of Compliance 71-9313*. This verification shall be performed by two independent individuals.

7.4.1.2 Review the maintenance records of the cask for situations where air may have leaked into the cask while it was in its storage configuration, i.e., while on the storage pad. If air has leaked into the cask while it was in its storage configuration, perform an evaluation prior to transportation of the fuel cladding for potential rod splitting due to exposure to an oxidizing atmosphere using the methodology given in ISG-22.

7.4.1.3 The assigned burnup loading value for each fuel assembly shall be obtained from a source controlled by the site's QA program and traceable to the TOTE or BURNUP output corresponding to when the fuel assembly was discharged from the reactor for the final time.

7.4.1.4 The assigned burnup loading value for each fuel assembly shall be from the TOTE or BURNUP computer codes. The value from these codes shall be reduced by a 1.04 factor to account for burnup uncertainties.

Note: The following steps may be performed at the ISFSI site. However, space and equipment requirements may require all operations to be performed at the plant loading area. The following steps permit either option.

A. Storage Area

7.4.1.5 Disconnect the overpressure system from the monitoring panel. Depressurize the overpressure tank and disconnect the tubing at the protective cover.

Note: The following 4 steps may not be necessary if preparation is done on the storage pad

7.4.1.6 Position the cask transporter over the cask.

7.4.1.7 Engage the lifting arms and lift the cask to the designated lift height.

7.4.1.8 Move the cask to the loading area.

7.4.1.9 Lower the cask down onto the floor, disconnect the cask transporter and remove the transporter from the loading area.

7.4.1.21 Lift the cask, and place the rear trunnions on the rear trunnion supports of the upending/downending frame.

7.4.1.22 Rotate the cask from the vertical to the horizontal position.

7.4.1.23 Using a spreader bar and lifting straps, lift the cask from the upending/downending frame and lower it onto the transport frame.

7.4.1.24 Install the tie-down straps.

7.4.1.25 Check if the surface dose rates and the surface contamination levels are within the regulatory limits. *Perform an external temperature survey for monitoring thermal performance.*

7.4.1.26 Prior to installing the impact limiters, inspect them visually for damage. The impact limiters may not be used without repair if any wood has been exposed. Damage due to handling other than small dings and scratches must be evaluated for their effect on the performance during the hypothetical drop and puncture accidents.

7.4.1.27 Install the top impact limiter spacer on the front end (lid end) of the cask and then remove the spacer lifting eye bolts.

7.4.1.28 Install the front (top) and the rear (bottom) impact limiters onto the cask. Lubricate the attachment bolts with Loctite N-5000 or an equivalent and torque to 60 - 80 ft-lb in the final pass.

7.4.1.29 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.4.1.30 Render the impact limiter lifting lugs inoperable by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.4.1.31 Install security seal on one tie-rod and lock sleeve.

7.4.1.32 Install the personnel barrier.

7.4.1.33 Check the temperature on all accessible surfaces to make sure that it is <185°F.

7.4.1.34 Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10 CFR 71.47 and 71.87.

7.4.1.35 Apply appropriate DOT labels and placards in accordance with 49 CFR 172. Prepare the final shipping documentation.

7.4.1.36 Release the loaded cask for shipment.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

The following reviews, inspections, and tests shall be performed on the TN-40 packaging prior to initial transport. Many of these tests will be performed at the fabricator's facility prior to delivery of the cask to the utility for use. Tests will be performed in accordance with written procedures approved by Transnuclear, Inc. For the TN-40 casks that have been fabricated, loaded and used for storage under 10CFR72 requirements, use of acceptance tests performed during their fabrication are also acceptable.

8.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility prior to initial use to ensure that the packaging conforms to the drawings and specifications. *The visual inspections include:*

- *cleanliness inspections,*
- *visual weld inspections as required by ASME Code [1],*
- *inspection of sealing surface finish, and*
- *dimensional inspections for conformance with the drawings included in Chapter 1 and in the Certificate of Compliance.*

The visual inspection includes verifying that all specified coatings are applied and the packaging is clean and free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its effectiveness. To the maximum extent practical, weld inspection is performed in accordance with the applicable ASME code sections [1]. Dimensions and tolerances shown on the drawings provided in Chapter 1 are confirmed by measurements. The sealing surfaces on the flange, lid and covers are inspected to ensure that there are no gouges, cracks or scratches that could result in an unacceptable leakage.

Prior to shipping, the packaging will be inspected to ensure that it is in good physical condition. This inspection shall include verification that all accessible cask surfaces are free of grease, oil or other contaminants, and that all cask components are in an acceptable condition for use.

8.1.2 Structural and Pressure Tests

The structural analyses performed on the packaging are presented in Chapter 2. To ensure that the packaging can perform its design function, the structural materials are chemically and physically tested to confirm that the required properties are met. To the maximum extent practical, welding is performed using qualified processes and qualified personnel, according to the ASME Boiler and the Pressure Vessel Code [1]. Base materials and welds are examined in accordance with the ASME Boiler and Pressure Vessel code requirements. NDE requirements for welds are specified on the drawings

provided in Chapter 1. All NDE is performed in accordance with written and approved procedures. The inspection personnel are qualified in accordance with SNT-TC-1A [2].

The containment welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB. Alternatives to the code taken regarding the containment vessel are described in Chapter 2, Section 2.11. The basket is designed, fabricated, and inspected in accordance with the ASME B&PV Code Subsection NB Alternatives to the code taken regarding the basket are described in Section 2.11. Welds of the noncontainment structure are inspected per the NDE acceptance criteria of ASME B&PV Code, Subsection NF.

The TN-40 fuel basket is designed, fabricated, and inspected in accordance with the ASME B&PV Code Subsection NB. Fusion weld tests as required are shown on drawings provided in drawing 10421-71-9.

The impact limiter attachment bolt material is tested to show the Charpy fracture toughness is at least 20 ft-lb at -20°F. The tie rod material is tested to show the Charpy impact test energy is at least 35 ft-lb at -20°F.

Pressure Tests

Prior to inital use a pressure test is performed on the cask assembly at a pressure of 25 psig. This is slightly higher than 1.5 times the maximum normal operating pressure of 15.7 psig. The test pressure is held for a minimum of 10 minutes. The test is performed in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 or NB-6300. All visible joints/surfaces are examined for possible leakage after application of the pressure. Temporary gaskets and seals may be used in place of the metallic seals during the test.

In addition, a bubble leak test is performed at a pressure of 3 - 5 psig on the neutron shield enclosure (outer shell, outer shell top and bottom rings). The purpose of this test is to identify any potential leak passages in the enclosure welds. The bubble leak test pressure is greater than the relief valve set pressure.

Load Tests

The lifting trunnions are designed to exceed 10CFR71.45(a) lifting requirements. A load test of 1.5 times the design lift load is applied to the trunnions for a period of ten minutes, to ensure that the trunnions can perform satisfactorily. *This has been approved previously in the NRR safety evaluations for the dry cask storage at Prairie Island [4].*

A force equal to 1.5 times the impact limiter weight will be applied to the lifting lugs of each limiter for a period of ten minutes. At the conclusion of the test, the impact limiter lifting lugs (including welds) will be:

a. Visually examined for defects and permanent deformations.

8.1.6 Neutron Absorber Tests

Boral® is the neutron absorber used for criticality control in the TN-40 basket. The neutron absorber plates may be monolithic, or they may consist of paired plates, one containing boron in the specified areal density, and the other composed of aluminum or aluminum alloy to make up the balance of the specified thickness and thermal conductance.

The TN-40 safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these materials.

The Boral® neutron absorber material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum. The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®.

8.1.7 Thermal Acceptance Tests

The thermal evaluation presented in Chapter 3 is based on design configurations and thermal properties taken from industry recognized standards for the specified materials. Therefore thermal acceptance tests of the TN-40 cask are not required.

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

The TN-40 cask *will* be used as a storage cask prior to use as a transport cask. If a loaded cask is taken from storage and prepared for transport, no load testing beyond the initial fabrication load test is required prior to shipment.

The leak test port (the overpressure port in the storage configuration) is closed by the overpressure transport cover with a single metallic seal. This flange and seal are not part of the containment boundary. The quick connect couplings in the vent and drain ports are not part of the containment boundary.

There are no valves or rupture discs on the TN-40 packaging containment.

8.2.4 Shielding

There are no periodic tests or inspections required for the TN-40 shielding. Radiation surveys will be performed of the package exterior to ensure that the limits specified in 10 CFR 71.47 are met prior to shipment.

8.2.5 Thermal

There are no periodic tests or inspections required for the TN-40 heat transfer components. *However, a thermal survey of the cask exterior will be performed prior to transport to ensure the thermal performance of the packaging.*

8.3 References

1. ASME Boiler and Pressure Vessel Code, Section III, 1989.
2. SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1987.
3. ANSI N14.5-1997, "Leakage Tests on Packages for Shipment of Radioactive Materials."
4. *Letter from Beth A. Wetzel (NRC) to Roger O. Anderson (NSPC), "Safety Evaluation and Safety Assessment Related to Dry Cask Storage at Prairie Island," June 12, 1995.*